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AQUEOUS PROCESSING OF LMFBR FUELS - TECHNICAL ASSESSMENT AND EXPERIMENTAL PROGRAM DEFINITION

Compiled and Edited by the Staff
of the
Chemical Technology Division

JUNE 1970

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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FOREWORD

This report was prepared in response to a request from the Reactor Development and Technology Division of the U.S. Atomic Energy Commission to perform a comprehensive assessment of the technical problems which would be encountered in extending the currently known aqueous reprocessing technology to fuels discharged from Liquid Metal Fast Breeder Reactors (LMFBR), and to define an experimental program for solving these problems which would be consistent with the overall LMFBR Program Plan, and more specifically with the Program Objectives and Plan of Action presented in Element 8, Fuel Recycle, (WASH-1108) under Task 8-3.1.

Because of the close interaction between processing, shipping, and waste management, the latter two areas were also included in the assessment and program planning. These also were to be made consistent with their respective Tasks 8-2.2 and 8-3.2 in the LMFBR Fuel Recycle Program Plan.

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AQUEOUS PROCESSING OF LMFBR FUELS - TECHNICAL ASSESSMENT AND EXPERIMENTAL PROGRAM DEFINITION

ABSTRACT

This report describes, in detail, the development program that has been initiated for the resolution of problems posed by the processing of spent fuels from Liquid Metal Fast Breeder Reactors. These fuels cannot be transported and processed by methods currently employed for light water reactor fuels owing to the high decay heat evolution and to the high plutonium content, which alters chemical requirements and necessitates considerations of criticality.

---

1. INTRODUCTION

Future Liquid Metal Fast Breeder Reactor (LMFBR) fuels cannot be processed economically by present LWR fuel processing practices. This is due to the increased plutonium content, specific power, and mechanical complexity of LMFBR fuels. Since the acceptance of the LMFBR concept is dependent on an economical fuel cycle, it is necessary to develop fuel reprocessing technology which will assure future reactor customers that processing will be available and that the cost will be acceptable. The proposed Aqueous Processing of LMFBR Fuel Program is intended to provide the technology to fill this need. The objectives of this program are: first, to reduce, with time, the uncertainties that exist in applying present aqueous processing technology to future LMFBR fuels; second, to provide the technology necessary to adapt LWR fuel processing plants to the processing of LMFBR fuels in the interim period before the LMFBR fuel load increases to the point that it can support a separate processing plant; and to provide the technology necessary for eventual commercial LMFBR fuel reprocessing plants. The latter, long-term approach is reserved for those problems that are occasioned by the economic incentive for short-decay processing and

that are of sufficient technical complexity to require an extensive development period.

It is expected that oxide fuels sheathed in stainless steel will be used in the early LMFBR's; for this reason, oxide fuels have been adopted as the reference feed material for initial processing studies. However, advanced fuels, such as the carbides and nitrides, are not being excluded from consideration. The core fuel element has three zones; the center is a core containing uranium oxide plus about 20% plutonium oxide as the fissionable material, and the two end sections, containing depleted uranium oxide, are the axial blanket sections of the element. While it is practicable to separate the core from the two end blanket sections and process them separately, there is no apparent advantage in doing so, and it has been assumed that the three sections will be processed together. Surrounding the core section of the reactor are elements of depleted uranium oxide that form the radial blanket. These also can be processed separately or mixed and processed simultaneously with the core fuel elements.

This report presents a comprehensive plan for carrying out a development program aimed at the application of present aqueous chemical processing technology to future LMFBR fuels. The status of each step of the fuel cycle from the reactor to fuel fabrication is analyzed. The problems that must be solved are discussed, and the development work needed to solve these problems is scheduled.

This program plan, which covers Task Areas 8-2 and 8-3 earlier defined by the LMFBR Program Office, describes the LMFBR fuels and the present state of aqueous processing practice. Process steps, along with evolutionary versions applicable to LMFBR processing requirements, are evaluated. The breakdown of tasks serves to delineate the proposed development program and covers a broad spectrum of parameters, such as fuel characteristics and cooling times, to ensure a generalized applicability to the various LMFBR concepts. In each task area, the work to be undertaken is discussed; the need is given; and the proposed approach is outlined.

It is our intent that this program plan will be updated and revised periodically to reflect the continuing progress made in the development of reactor fuels and in this processing program.

2. SUMMARY

Nearly every major fuel processing facility in existence today employs the Purex solvent extraction process. The outstanding favorable features of this process - unexcelled separation efficiency, versatility, ease of adaptation to continuous high-capacity throughput - and the vast operating experience available from major processing facilities make it an obvious choice for adaptation to LMFBR fuel processing.

The average burnup of the fast breeder reactor is expected to be on the order of 80,000 Mwd/ton for the core and about 33,000 Mwd/ton for the mixed core and blanket (see Table 2.1). This compares with 34,000 Mwd/ton proposed for LWR oxide fuels. Fast breeder reactors will have a high plutonium inventory and a high specific power (about 150 kw per kilogram of heavy metal oxide in the core as compared with about 35 kw/kg for LWR's). This combination of high burnup and high specific power makes the decay heat of discharged core fuel from fast breeder reactors about a factor of 5 higher than that of thermal reactors. There will also be a higher concentration of long-lived fission products in the LMFBR core fuel, roughly in proportion to the higher burnup. Fast breeder fuels have a higher plutonium content than LWR fuels by about a factor of 10, which aggravates the concern over criticality, emphasizes the importance of plutonium chemistry, and may occasion the need for shielding for plutonium product handling operations to attenuate the neutrons generated by (α, n) reactions and spontaneous fissions. The value of the plutonium content of LMFBR fuel results in an economic incentive for minimizing the decay period before reprocessing.

During the interim period, before facilities designed specifically for short-cooled LMFBR fuel are available, it should be possible to process the fuel from early fast breeder reactors, the FFTF, and demonstration

Table 2.1. Fuel Comparison: Light-Water Reactors and Future Fast Breeders

	Typical Fast Reactor Fuel [a]	LWR Fuel, UO_2-PuO_2 (with Pu Recycle) [b]	LWR Fuel, UO_2 [c]
Burnup, Mwd/ton			
Core	80,000	33,900	20,300
Core and blanket, average	33,000		
Specific power, kw/kg fuel in core (at 85% load factor)	148.8	33.76	33.61
Decay heat, w/kg fuel			
30-day decay:			
Core	199	50.7	46.6
Core and blanket	82		
150-day decay:			
Core	76.4	21.5	16.9
Core and blanket	28		
Plutonium content, g/kg fuel			
Core	243	26.9	7.5
Core and blanket; 72% Pu is fissile	104		

Table 2.1. (Cont'd.)

	Typical Fast Reactor Fuel [a]		LWR Fuel, UO_2-PuO_2 (with Pu Recycle) [b]	LWR Fuel, UO_2 [c]
Iodine, curies/ton after cooling period	30-Day	150-Day	150-Day	150-Day
Core	3.7×10^5	11.92	2.8	2.57
Core and blanket	1.5×10^5	4.92		
Process containment factor for 1-ton/day capacity	$\sim 10^8$	$\sim 10^3$	$\sim 10^2$	$\sim 10^2$

[a] Reactor No. 21 (LMFBR; AI reference oxide). See Fuel Recycle Task Force Report, WASH-1099 (in preparation).

[b] Reactor No. 16 (LWR; Pu recycle with tails). See Fuel Recycle Task Force Report, WASH-1099 (in preparation).

[c] Reactor No. 1 (LWR; UO_2). See Fuel Recycle Task Force Report, WASH-1099 (in preparation).

reactors in existing LWR processing facilities modified for this purpose. Such fuel could be cooled about 150 days (not necessarily optimum for commercial fast breeder fuel but economically tolerable during the interim period) and then processed at about one-half nominal capacity in an existing LWR plant altered to accommodate the greater length of the FFTF elements, deal with sodium contamination, and provide capacity for the higher plutonium content.

The long-range economic success of the LMFBR concept is dependent upon a relatively cheap fuel cycle, and it seems reasonably important that a reliable knowledge of chemical processing feasibility and costs exist at the time that the LMFBR concept is to be attractive to commercial interests on the basis of its favorable economics. The economic advantage of large-scale processing plants is so prominent that central processing facilities with throughputs on the order of 5 tons/day or greater are surely to be preferred over a proliferation of small (< 1 ton/day) facilities, and the importance of safe, economical transport of fuel to central processing facilities has been recognized. The continued expansion of the power economy and the increasing technology of power transmission make large central power parks, on the order of 20,000 Mw, a plausible concept. An "on-site" processing facility of corresponding capacity (ca. 1 ton/day) begins to acquire the favorable economic characteristics of the large central processing facility. The essential processing problems are similar, however, and generalized process development studies will be readily applicable to the economic analysis on which reactor designers base the selection of the optimal fuel cycle concept. The process development program should provide the technical foundation for the design and operation of the processing facilities of probable interest (capacity, preprocessing decay, etc.). This program will include consideration of a spectrum of LMFBR fuels. Fortunately, the fuel designs of the FFTF and the five follow-on reactors are superficially similar; that is, each is approximately 5 in. by 18 ft long, consists of about 200 stainless-steel-sheathed $(Pu,U)O_2$ elements about 1/4 in. in diameter, is spaced with wire wrapping or grids, and is encased in a 1/8-in. stainless steel shroud assembled by welding to end fittings.

Reactor fuel has been shipped safely in casks for a number of years. However, the problem of heat dissipation from short-cooled LMFBR fuel is much more severe, and it is apparent that a more refined technology must be exploited in the design of LMFBR shipping casks. Steel rather than lead may be used for shielding, not only for its greater strength but because it can safely endure higher temperatures under accident conditions. More sophisticated heat transporters can provide greater cooling capacity than the simple conduction devices now used. Mechanically reliable closures that can probably be adapted to survive any of the postulated impact conditions of a major accident are now commercially available. The success of these studies may enable fast breeder fuel to be shipped both safely and economically, probably at a unit cost (mills/kwhr of electricity) approximating that for shipping LWR fuels at the present time.

After transport, the mechanical disassembly of an LMFBR irradiated fuel element is a formidable problem because of the need for efficient dissipation of large amounts of heat while performing remote operations. A variety of head-end methods will be assessed for this initial important step. A finite probability exists that failed fuel pins, containing sodium coolant in the interstices of oxide fuel (sodium-logged), will be present.

The retention of ^{131}I, encountered even in the processing of 150-day-cooled LWR fuels, will be an overriding problem with LMFBR fuels. Iodine retention factors in the range of 10^3 (near the range of present practice) to 10^8 will be required for 150-day-cooled FFTF fuel and for future 30-day-cooled LMFBR fuel, respectively. The chemistry of iodine is complex, and there is not sufficient knowledge of the behavior of iodine in the various conditions encountered in a fuel processing plant to design a treatment system with the reliability required for LMFBR fuels. For example, the presence of organic vapors and organic iodides in the off-gas system reduces the efficiency of iodine removal steps. Early studies indicate that the organic materials can be eliminated by conversion to CO_2, H_2O, and I_2 in a catalytic oxidizer using Hopcalite catalyst (oxides of copper and manganese) at 150 to 500°C. Iodine

retention was increased by at least two orders of magnitude in a field test of a catalytic oxidizer--charcoal adsorber system in the Transuranium Processing Plant at ORNL

Fission product gases interstitially trapped in oxide fuels can be volatilized and collected in a relatively small volume when these fuels are heated in oxygen at temperatures greater than about 400°C. Similar volatilization is expected on oxidation of carbide or nitride fuels. Early studies show that nearly quantitative removal of noble gases and approximately 90% removal of iodine and tritium are obtained from UO_2 but that removal is more difficult and unpredictable for 20% PuO_2--80% UO_2. Tritium removal was greater than 90%, whereas noble gas removals varied from 20 to 98%; the removal of iodine is under study. High-efficiency retention of the iodine is enhanced with increased concentration, and the removal of the active iodine after the fuel is sheared but before it reaches the dissolver would greatly reduce the problems attributed to iodine in current processing experience. The oxidative heat treatment may have other advantages as well. For example, the sodium that is present as a result of failed fuel rods will be deactivated in this step. In addition, the oxidation of UO_2 to U_3O_8 involves a phase change which tends to degrade the oxide into a fine granular form. The phase change, in turn, makes the physical separation of the uranium oxide from the stainless steel cladding a possibility and provides for a faster dissolution of the uranium oxide in nitric acid. This advantage is probable but whether it applies also to the mixed oxide is yet to be determined. The oxidation of uranium oxide before dissolution reduces the quantity of nitrous oxides evolved in the dissolver off-gas.

Radiation damage to the solvent, in spite of the high specific activity of the LMFBR fuels, should not be a significant problem. By using high-speed contactors, the extent of solvent exposure to radiation can be held far below the limits that have already proved to be satisfactory in plant practice. These contactors also have high capacity and quick response and are amenable to scale-up to large capacity. On the bases of laboratory and hot-cell tests, we have concluded further

that pulsed columns should be usable without serious consequence to solvent performance. Confirmatory tests are continuing.

Past experience with the difficulty of interpreting anomalous behavior when the extrapolation of laboratory data to plant application is undertaken has led us to believe that radioactive engineering-scale experiments to remove scale-up uncertainties may be justified. Planning for the engineering-scale experiments for this purpose could begin in mid-1971, with equipment installation and check-out and operation spanning the period 1972 to 1974. The culmination of such engineering tests should include operation under full-scale radioactivity, probably with FFTF fuel, which should be available about that time (FFTF is scheduled to go critical in 1973). The assurance of LMFBR technology will have been established during the initial growth of the LMFBR power plants, giving a lead time of approximately five to seven years for commercial fuel processing interests to complete the site selection, design, construction, and startup of a plant specifically designed to accommodate fast reactor fuel in the early 1980's.

The concepts of flowsheet (or equipment) that dictate plant design must be established before meaningful engineering tests can materialize. These tasks are scheduled early to establish that a concept has economic merit and technical reliability and is suitable for a design basis. Preliminary design can then proceed apace with the continuing development of detailed technical information and parameters. Later, with operation of the engineering-scale experiments, process support and technical backup for resolving the anomalous problems that invariably arise during the early phases of full-scale operation will be required.

In deciding the priority of the various tasks and subtasks in the period FY 1969 to 1973, we have assigned first priority to reducing the uncertainties that exist in applying present aqueous technology to LMFBR fuels (see Table 2.2). The major problems involved in this category are safe transport, heat dissipation during handling, effect of criticality on plant scale-up, and off-gas retention, especially iodine. In addition, all the chemical processing steps will be examined in scouting experiments and subjected to testing in order to establish

Table 2.2. Identification of Top Priority Tasks

Task	Period		
	1969-1973	1971-1974	1974-1977
Shipping			
Heat dissipation	Priority 1		
Containment	Priority 1		
Receiving and storage			
Sodium removal	Priority 1		
Heat dissipation	Priority 1		
Head-end processing			
Heat dissipation	Priority 1		
Volatile fission product removal	Priority 1		
Leaching	Priority 2		
Feed preparation	Priority 2		
Solvent extraction	Priority 2		
Pu isolation	Priority 2		
Waste disposal			
Gaseous fission products	Priority 1		
Other	Priority 2		
Off-gas treatment	Priority 1		
Engineering-scale experiments			
Design		Priority 1	
Construction		Priority 1	
Operation and support			Priority 1

the technology necessary for subsequent commercial LMFBR processing plants. Engineering-scale experiment design and construction are scheduled to begin in mid-1971 and in 1972, respectively; thus these tasks are given first priority in the 1971-74 period. Beyond 1974, first priority shifts to the operation of the engineering-scale experiments and the supporting development work.

3. GENERAL ASPECTS OF LMFBR FUEL PROCESSING

The development of aqueous processes for the recovery of plutonium from spent LMFBR fuels requires the solution of many challenging problems. As compared with light-water reactor (LWR) fuels, the LMFBR core fuels are characterized by high specific power, high burnup, and high plutonium content.[1]

The high specific powers result in high heat removal requirements for the fuel rods, which, in turn, require small rod diameters (~0.25 in.) and possibly a sodium bond between the fuel and cladding. Even if the LMFBR elements are not fabricated with a sodium bond, it is likely that a significant fraction of the rods will contain sodium as a result of cladding failures. Schemes must be devised for removal of this sodium prior to fuel element dissolution because sodium can react explosively with fuel dissolvents.

The effect of the requirements for withstanding high burnups and for compatibility with sodium at a temperature of up to 1300°F is to focus attention on fuel cladding materials other than Zircaloy and aluminum, for which most processing experience is available. Vanadium, nickel, and ferrous alloys, as well as the austenitic stainless steels, are being considered. The various cladding materials that may evolve for LMFBR's dictate a versatile mechanical scheme for exposure of fuel to dissolvent as opposed to chemical methods.

The high plutonium content of the fuel and the attendant high inventory charge provide incentive to reduce the postirradiation decay (i.e., cooling) time. Short cooling periods increase the problems of

heat removal in shipping, handling, and processing, and require greatly enhanced schemes for treatment of short-lived volatile fission products. The high plutonium content also dictates the use of advanced schemes for criticality control since only a very small unit of fuel would be required to initiate criticality in the large vessels that are customarily used for processing of low-enriched LWR fuels.

The following sections present a more detailed description of LMFBR fuel characteristics, with emphasis on the near-term fuel for which design concepts are available. Section 3.3 reviews current technology for aqueous processing of LWR fuels and identifies, in a general way, options for the aqueous processing of LMFBR fuels on interim and long-term bases.

3.1 LMFBR Fuel Element Properties

Because of their relatively advanced technology, mixed oxide fuel (UO_2-PuO_2) and austenitic stainless steel cladding have been chosen for the Fast Flux Test Facility and will probably be used in first cores of demonstration reactors. Later fuels may include carbides, nitrides, and metal alloys. Later cladding materials may include alloys of vanadium (e.g., V-10Ti-Cr-Mo-Nb), nickel (e.g., Ni-20Cr-Fe-Nb-Mo-Ti), and cobalt (e.g., Co-20Cr-Fe-Ni-Mo-Nb).[2]

Table 3.1 presents a comparison of properties that will probably encompass the scope of the early LMFBR's. These data include those for the FFTF;[3,4] the Reference Oxide LMFBR design that was developed by Atomics International (AI) for the Systems Analysis Task Force Study;[5] and conceptual 1000-Mw (electrical) LMFBR designs developed by AI,[6-9] Babcock and Wilcox,[10,11] Combustion Engineering,[12-14] General Electric,[15-17] and Westinghouse [18-20] in the follow-on studies made under the direction of the Argonne National Laboratory.

All of these reactor concepts utilize austenitic stainless steel cladding; five use oxide fuel and a gas bond and two use carbide fuel and a sodium bond. The core average specific power varies from 100

Table 3.1. Comparison of Properties of FFTF and Conceptual LMFBR's

	FFTF	AI Ref. Oxide	Follow-On Plants[a]				
			AI	B&W	CE	GE	W
Fuel form	Oxide	Oxide	Oxide	Oxide	Carbide	Oxide	Carbide
Nuclear data (at full power)							
Power, Mw(thermal)	400	2500	2400	2450	2465	2417	2600
Core							
Avg. sp. pwr., Mw/metric ton (U + Pu)	158	175	115	103	122	157	161
Burnup, Mwd/metric ton	45,000	80,000	67,400	100,000	100,000	100,000	100,000
Charge, metric ton (U + Pu)	2.5	12.6	19.1	23.9	17.32	13.45	13.7
Charge, metric ton (fissile Pu)	0.51	1.96	2.33	2.77	1.92	1.62	1.98
Charge, Pu/(U + Pu)	0.23	0.22	0.171	0.16	0.16	0.17	0.18
Rod avg. linear pwr., kw/ft	7.8	10.0	10.1	8.3	29.5	9.63	18.5
Rod max. linear pwr., kw/ft	13.1	16.	16.0	14.4	43.5	16.3	39.2
Max. clad. temp., °F		1230	1220	1174	1170	1300	1300
Max. fuel temp., °F	4200	4748		4000	2310		2620
Axial blanket							
Avg. sp. pwr., Mw/metric ton (U)		5.5	5.71	2.83	10.2	12.2	
Burnup, Mwd/metric ton (U)		2500	3340	2000.	7800	7200	
Charge, metric ton (U)		7.32	13.1	19.1	25.97	15.0	11.
Radial blanket							
Avg. sp. pwr., Mw/metric ton (U)		10.	7.89	8.38	24.2	6.95	3.0
Burnup, Mwd/metric ton (U)		8100	12,700	9000.	57,000	7000	
Charge, metric ton (U)		26.7	15.6	16.1	5.69	16.7	33.
Core-axial blanket fuel assembly	Hex.	Hex.	Hex.	Hex.	Circular	Hex.	Square
Number per reactor	76	252	274	288	219	265	244

Table 3.1. (Cont'd.)

	FFTF	AI Ref. Oxide	Follow-On Plants[a]				
			AI	B&W	CE	GE	W
Shroud material	SS	SS	SS	SS	SS	SS	SS
Thickness, in.	0.140	0.168	0.140	0.128	0.050	0.060	0.020
Length							
Core, in.	36	48	42.8	34.7	24	30	39.3
Axial blanket (each end), in.		12	12,15	14	18	15	15
Plenum, in.	36	60	38		18	20	25
Rod, in.	87	144	110	66	79.8	82.3	100
Overall, in.	168	212	180	110	144	119	122.5
Width or diameter, in.	4.615	5.4	5.62	6.45	6.53	5.62	5.25
Fuel rod data							
Number per assembly	217	217	217	331	163	331	168
Outside diameter, in.	0.230	0.25	0.300	0.280	0.400	0.25	0.302
Pitch,[c] in.	0.286	0.341	0.359	0.337	0.451	0.30	0.375
Clad thickness, in.	0.015	0.015	0.0175	0.010	0.011	0.010	0.012
Clad material	316 SS	304 SS	316 SS	304 SS	316 SS	316 SS	316 SS
Fuel density, % theoretical	88	80, 93	88, 93	85	99	85	90
Bond	He	He	He	He	0.014[b] Na	He	0.03[b] Na
Vented?	No	No	No	Yes	Yes	Yes	Yes
Type of spacer	Wire	Wire	Wire	Ribbon	Wire	Wire	6 grids
Spacer material	SS	SS	SS	SS	SS	SS	Inconel
Radial blanket assembly		Hex.	Hex.	Hex.	Hex.	Hex.	Square
Number per reactor		147, 87	66, 72	138	60	126	273
Length							
Fuel, in.		60, 48	56, 43	42.7	36	60	69.3
Rod, in.		84, 72	66, 53	66	51.8	82	100
Overall, in.		212	180	110	144	119	122.5

Table 3.1. (Cont'd.)

	FFTF	AI Ref. Oxide	Follow-On Plants[a]				
			AI	B&W	CE	GE	W
Width or diameter, in.		5.4	5.62	6.45	6.53	5.62	5.25
Fuel rod data							
Number per assembly		169, 91	91, 61	91	91	127	60
Outside diameter, in.		0.35, 0.51	0.50, 0.62	0.610	0.600	0.42	0.55
Pitch,[c] in.		0.36, 0.49	0.55, 0.68	0.640	0.652	0.47	0.604
Clad thickness, in.		0.015	0.0175	0.017	0.020	0.015	0.012
Clad material		304 SS	316 SS	304 SS	316 SS	316 SS	316 SS
Fuel density, % theoretical		93	93	90	99	95	Oxide,95
Bond		He	He	He	0.02[b] Na	He	He
Vented?		No	No	Yes	No	No	No
Type of spacer		Wire	Wire	Ribbon	Wire	Wire	
Spacer material		SS	SS	SS	SS	SS	

[a] AI = Atomics International; B&W = Babcock and Wilcox; CE = Combustion Engineering; GE = General Electric; W = Westinghouse.

[b] Inches.

[c] The Westinghouse rods have a square pitch; all others have a triangular pitch.

to 175 Mw/metric ton, and the average core burnup varies from 45,000 to 100,000 Mwd/metric ton. The average linear power and the peak linear power in the core fuel rods are 8 to 10 and 14 to 16 kw/ft, respectively, in the oxide designs, and 18 to 30 and 39 to 44 kw/ft, respectively, in the carbide designs. The 1000-Mw (electrical) designs employ an inner array of integral core-axial blanket fuel assemblies surrounded by a layer of axial blanket assemblies. The blankets are characterized by a specific power of about 10 Mw/metric ton and a burnup of approximately 10,000 Mwd/metric ton.

3.1.1 Overall Description of Fuel Elements

The core-axial blanket fuel elements of the oxide conceptual designs are characterized by a (normally hexagonal) bundle of either 217 or 331 fuel rods. The carbide designs have fewer rods of larger diameter. The overall width of the assembly ranges from 4.6 to 6.5 in., and the overall length ranges from 9 to 18 ft. The most common fuel rod spacer is a stainless steel spiral wire wrap with an amplitude of 6 to 12 in., although some current designs employ thin grids; the Westinghouse design utilizes six grids of Inconel 718. The shrouds for which data are available are made of stainless steel of thicknesses varying from 0.05 to 0.168 in.

The radial blanket fuel assemblies generally have the same overall size and outward appearance as the core-axial blanket assemblies. The rods are generally larger in diameter and fewer in number.

3.1.2 Fuel Rod Description

The core-axial blanket fuel rods in the oxide designs have outside diameters varying from 0.23 to 0.30 in. and wall thicknesses varying from 0.010 to 0.0175 in. The two carbide designs have rod diameters of 0.30 and 0.40 in. and a cladding wall thickness of about 0.010 in. The length of the core section varies from 24 to 48 in., and the length of the axial blanket section (each end) varies from 12 to 18 in. The designs with nonvented rods utilize a gas plenum with a length up to

60 in. The rod lengths vary from 66 to 144 in. The radial blanket rods have outside diameters ranging from 0.35 to 0.62 in.

3.1.3 Fuel and Cladding

The smeared density of the fuel in the oxide core designs ranges from 80 to 88% of theoretical, while the carbide core fuels have densities up to 99% of theoretical. The candidate fuel forms are pellets or "rods" of mechanically blended or coprecipitated oxides, vibratorily compacted sol-gel oxides, and pelleted carbides. The fuel in the blankets is generally more dense than that in the core because of the lesser requirement for void space to compensate for swelling.

The cladding material for FFTF and all of the conceptual designs is either 304 or 316 (2.5 wt % of molybdenum) stainless steel.

3.2 Properties of Irradiated Fuel

The nuclear properties of the conceptual LMFBR fuels shown in Table 3.1 are sufficiently similar that each type of spent fuel could contain essentially the same isotopic concentrations and generate equivalent thermal power following normal reactor operating conditions that would include variable load factor and, possibly, early refueling. For this reason, this section will present detailed spent fuel properties for only one of the concepts, based on irradiation conditions that will provide probable upper limit concentrations of fission products and thermal power as a function of postirradiation decay time. The AI Reference Oxide concept was selected for this analysis because it has the highest specific power, and relatively complete documentation is presently available.

Masses, radioactivity, and thermal power of fission products, actinides, and activation products in the AI Reference Oxide Reactor were calculated as a function of irradiation and cooling time, using the ORIGEN program and the library of nuclear data (Appendix D) that are discussed in Sect. 4.12. The core was assumed to operate at a

constant average specific power of 148.15 Mw/metric ton for 540 days (equivalent to a load factor of 0.85). The specific power of the axial blanket was input as a step function varying from 2.27 Mw/metric ton initially to 6.99 Mw/metric ton at discharge (540 days), and averaging 4.63 Mw/metric ton. The specific power of the radial blanket was varied from 2.32 Mw/metric ton initially to 14.38 Mw/metric ton at the 540-day discharge time, and averaged 8.4 Mw/metric ton. Transient concentrations of the approximately 700 nuclides in the current data library were calculated. Sections 3.2.1-3.2.3 present summary results of the most important nuclides in the spent fuel in terms of mass, radioactivity, and thermal power. All results are based on one metric ton of uranium, or uranium plus plutonium, as charged to the core, axial blanket, or radial blanket. The plutonium charged to the core was assumed to have the typical isotopic composition produced by LWR's.

Results are also presented for blends of core and blanket fuel. The core-axial blanket blend is illustrative of the mixture of nuclides that would result from dissolution of core-axial blanket fuel elements. The core-axial blanket-radial blanket blend may be used for estimation of the properties of the mixed waste generated by fuel processing. Fuels from the three regions are blended in the following proportions, by weight: 0.3624 of core, 0.2106 of axial blanket, and 0.4269 of radial blanket.

3.2.1 Fission Products

Table 3.2 and the tables of Appendix A present calculated masses, radioactivity, and thermal power of important fission products in spent LMFBR core, axial blanket, and radial blanket as a function of post-irradiation cooling time. Similar calculations made for core fuel irradiated to a burnup of 100,000 Mwd/metric ton at a specific power of 157 Mw/metric ton and a load factor of 80% resulted in up to 20% higher concentrations of some of the long-lived fission products, such as ^{85}Kr and ^{90}Sr, but lower gross thermal power by 5 to 15%.

Table 3.2. Fission Product Activity of Spent LMFBR Core Fuel as a Function of Decay Time

Burnup = 80,000 Mwd/metric ton

Specific power = 148.15 Mw/metric ton

Nuclide activities are given in curies per metric ton (U + Pu) charged to core

	On Discharge from Reactor	After Cooling Times of:				
		30 days	90 days	150 days	3 years	30 years
^{3}H	2,340	2,330	2,310	2,290	1,980	432
^{85}Kr	24,900	24,800	24,500	24,300	20,600	3,610
^{86}Rb	8,150	2,670	288	31		
^{89}Sr	2,240,000	1,500,000	674,000	303,000	1.01	
$^{90}Sr + ^{90}Y$	209,000	206,000	204,000	204,000	191,000	98,200
^{91}Y	3,080,000	2,170,000	1,070,000	527,000	7.56	
^{95}Zr	7,050,000	5,120,000	2,700,000	1,420,000	59.2	
^{95m}Nb	141,000	109,000	57,300	30,200	1.26	
^{95}Nb	6,970,000	6,510,000	4,460,000	2,660,000	126	
$^{99}Mo + ^{99m}Tc$	13,950,000	8,510	0.00289			
$^{103}Ru + ^{103m}Rh$	14,480,000	8,570,000	3,000,000	1,049,000	0.0675	
$^{106}Ru + ^{106}Rh$	6,820,000	6,460,000	5,760,000	5,140,000	862,000	0.00698
^{110m}Ag	4,650	4,280	3,630	3,080	231	
^{110}Ag	323,000	557	472	401	300	
^{111}Ag	505,000	31,600	123	0.482		
^{113m}Cd	341	340	337	334	294	77.3
^{115m}Cd	954	558	224	85.0		
^{119m}Sn	49.9	46.0	38.9	32.9	2.39	
^{121m}Sn	142	142	142	142	138	108
^{123m}Sn	2,070	1,750	1,250	900	4.74	
^{125}Sn	153,000	16,700	200	2.40		
^{125}Sb	50,500	50,700	48,800	46,700	24,000	23.4
^{125m}Te	17,100	17,900	18,700	18,600	9,960	9.7
^{126}Sb	10,200	1,940	73.8	6.85	4.36	4.36
^{127}Sb	846,000	4,040	0.0888			
$^{127m}Te + ^{127}Te$	1,023,000	315,000	213,000	145,600	354	
^{129m}Te	818,000	446,000	131,000	38,600		
^{129}Te	1,960,000	286,000	84,100	24,800		
^{129}I	0.129	0.131	0.132	0.133	0.133	0.133
^{131}I	4,320,000	336,000	1,920	10.9		
^{131m}Xe	45,100	15,100	581	17.9		
$^{132}Te + ^{132}I$	12,140,000	20,500	0.0583			
^{133}Xe	7,920,000	181,000	67.5	0.0252		
^{134}Cs	78,600	76,500	72,400	68,400	28,500	3.16
^{136}Cs	365,000	73,800	3,010	123		
$^{137}Cs + ^{137m}Ba$	515,000	514,000	512,000	510,000	480,000	257,000
$^{140}Ba + ^{140}La$	12,960,000	2,690,000	104,500	4,060		
^{141}Ce	6,760,000	3,580,000	991,000	275,000		
^{143}Pr	6,380,000	1,850,000	89,000	4,270		
$^{144}Ce + ^{144}Pr$	6,820,000	6,180,000	5,340,000	4,620,000	458,000	
^{147}Nd	2,830,000	434,000	10,300	242		
^{147}Pm	812,000	822,000	792,000	758,000	382,000	302
^{148m}Pm	177,000	108,000	40,000	14,900		

Table 3.2 (continued)

	On Discharge from Reactor	After Cooling Times of:				
		30 days	90 days	150 days	3 years	30 years
^{151}Sm	10,900	10,900	10,900	10,800	10,600	8,620
^{154}Eu	2,530	2,520	2,500	2,490	2,220	690
^{155}Eu	209,000	202,000	190,000	178,000	66,200	2.14
^{156}Eu	304,000	77,600	4,850	303		
^{160}Tb	34,100	25,600	14,400	8,070	0.906	
^{161}Tb	47,400	2,330	5.61	0.0135		
$^{162}Gd + {}^{162m}Tb$	24,200	22,800	20,400	18,140	3,020	
Total curies/metric ton	621,000,000	48,800,000	26,800,000	18,200,000	2,540,000	370,000
Beta power, w/metric ton	4,140,000	100,000	62,100	48,700	6,980	586
Gamma power, w/metric ton	2,210,000	95,600	43,900	25,700	1,840	494

3.2.2 Actinides

Tables 3.3 and 3.4 and the tables of Appendix B present calculated masses, activity, and thermal power of actinides in the core, axial blanket, and radial blanket. The yields and isotopic compositions of the isotopes ^{239}Pu through ^{242}Pu are generally in good agreement with those that have been predicted in reactor design studies. The ^{238}Pu content of the plutonium is predicted to decrease from 1.2% in the core charge to 0.93% in the discharged core. A ^{236}Pu concentration of 0.0092 ppm in the discharged plutonium is predicted.

3.2.3 Activation Products

Table 3.5 and the tables of Appendix C present calculated concentrations, radioactivity, and thermal power of the activation products of the oxygen, stainless steel, and sodium charged to the core. These data include only the stainless steel cladding in the core zone and assume a 0.001-in.-thick layer of sodium at the fuel-cladding interface.

3.2.4 Effect of Irradiation of Fuel Elements

Mechanical and chemical properties of irradiated LMFBR core fuel are presently conjectural because fuels have not been irradiated to the desired LMFBR exposure under conditions that include a linear power of 15 to 20 kw/ft, approximately 10% fission of total heavy atoms, a fast fluence of about 5×10^{23} neutron/cm^2, and a maximum fuel temperature in the range of 4000 to 4800°F.[21] Only a few general statements can be made that will guide the planning of schemes for transport and handling of the fuel elements.

Fuel. - The principal effect of irradiation on the fuel is swelling caused by gaseous and solid fission products. The planned use of oxide fuel with a low smeared density and a cladding material with relatively high strength may combine to produce relatively little change in the outer fuel dimensions, but some experiments indicate volumetric expansions of several percent regardless of the internal porosity. Fission gas releases of more than 80% have been observed in oxide fuels under conditions

Table 3.3. Isotopic Composition of Actinides in Spent LMFBR Core Fuel as a Function of Decay Time

Burnup = 80,000 Mwd/metric ton
Specific power = 148.15 Mw/metric ton
Nuclide concentrations are given in grams per metric ton (U + Pu) charged to core

	As Charged to Reactor	On Discharge from Reactor	After Cooling Times of:				
			30 days	90 days	150 days	3 years	30 years
^{4}He	0	1.66	1.82	2.10	2.33	4.10	20.5
^{233}U	0	0.00122	0.00122	0.00122	0.00122	0.00122	0.00122
^{234}U	0	21.8	22.9	25.2	27.6	64.2	407
^{235}U	0	6.18	6.44	6.98	7.51	15.9	104
^{236}U	0	8.72	9.15	10.0	10.9	24.6	167
^{238}U	782,000	719,000	719,000	719,000	719,000	719,000	719,000
^{237}Np	0	176	179	180	180	190	555
^{236}Pu	0	0.00178	0.00175	0.00168	0.00162	0.00086	0.0000012
^{238}Pu	2,590	1,820	1,830	1,840	1,840	1,840	1,490
^{239}Pu	130,000	116,000	117,000	117,000	117,000	117,000	116,000
^{240}Pu	51,800	52,400	52,400	52,400	52,400	52,400	52,300
^{241}Pu	26,000	14,500	14,500	14,400	14,200	12,400	2,940
^{242}Pu	8,760	9,020	9,020	9,020	9,020	9,020	9,020
^{241}Am	0	1,270	1,330	1,460	1,590	3,410	12,500
^{242}Am	0	24.6	24.6	24.6	24.6	24.3	21.5
^{243}Am	0	710	711	711	711	710	709
^{242}Cm	0	61.6	54.6	42.3	32.8	0.645	0.0517
^{243}Cm	0	2.32	2.32	2.31	2.30	2.17	1.21
^{244}Cm	0	42.3	42.2	41.9	41.7	37.7	13.4
Total	1,000,000	915,000	915,000	915,000	915,000	915,000	915,000

Table 3.4. Radioactivity of Actinides in Spent LMFBR Core Fuel as a Function of Decay Time

Burnup = 80,000 Mwd/metric ton
Specific power = 148.15 Mw/metric ton
Nuclide activities are given in curies per metric ton (U + Pu) charged to core

	Charged to Reactor	Discharged from Reactor	After Cooling Times of:				
			30 days	90 days	150 days	3 years	30 years
^{228}Th + daughters	0	0.00519	0.0060	0.00816	0.0107	0.0744	0.202
^{232}U	0	0.00595	0.00668	0.00812	0.00949	0.0249	0.0337
^{234}U	0	0.135	0.142	0.156	0.171	0.397	2.52
^{237}U	0	302,000	13,900	29.2	0.0617		
^{238}U	0.260	0.239	0.239	0.239	0.239	0.239	0.239
^{237}Np	0	0.124	0.127	0.127	0.127	0.134	0.391
^{239}Np	0	81,500,000	11,900	137	137	137	136
^{236}Pu	0	0.946	0.931	0.894	0.859	0.458	0.000643
^{238}Pu	43,800	30,700	30,800	31,000	31,100	31,000	25,200
^{239}Pu	7,940	7,120	7,150	7,150	7,150	7,150	7,140
^{240}Pu	11,400	11,600	11,600	11,600	11,600	11,500	11,500
^{241}Pu	2,970,000	1,660,000	1,650,000	1,640,000	1,620,000	1,420,000	325,000
^{242}Pu	34.2	35.2	35.2	35.2	35.2	35.2	35.2
^{241}Am	0	4,120	4,320	4,730	5,140	11,100	40,500
^{242m}Am	0	239	239	239	239	236	209
^{242}Am	0	393,000	239	239	239	236	209
^{243}Am	0	137	137	137	137	137	136
^{242}Cm	0	204,000	181,000	140,000	109,000	2,130	171
^{243}Cm	0	107	107	106	106	100	55.7
^{244}Cm	0	3,430	3,420	3,400	3,380	3,060	1,090
Total	3,030,000	167,000,000	1,920,000	1,840,000	1,790,000	1,480,000	422,000

Table 3.5. Cladding Activity of Spent LMFBR Core Fuel as a Function of Decay Time

Burnup = 80,000 Mwd/metric ton
Specific power = 148.15 Mw/metric ton
Nuclide activities are given in curies per metric ton (U + Pu) charged to core

	On Discharge from Reactor	After Cooling Times of:				
		30 days	90 days	150 days	3 years	30 years
^{14}C	0.0584	0.0584	0.0584	0.0584	0.0583	0.0581
^{32}P	2,310	539	29.4	1.61		
^{33}P	13.5	5.86	1.11	0.210		
^{51}Cr	61,700	29,200	6,550	1,470		
^{54}Mn	150,000	140,000	122,000	107,000	12,200	
^{55}Fe	76,000	74,300	71,200	68,100	34,100	25.5
^{59}Fe	10,300	6,520	2,590	1,030		
^{58}Co	410,000	307,000	171,000	95,500	10.00	
^{60}Co	1,350	1,330	1,310	1,280	908	25.9
^{59}Ni	2.13	2.13	2.13	2.13	2.13	2.13
^{63}Ni	67.8	67.7	67.6	67.6	66.3	54.1
Total	1,270,000	559,000	375,000	274,000	47,400	108

less stringent than those expected in LMFBR's. Carbide fuels are expected to retain their fission product gases under the temperature conditions of interest but may suddenly release these gases in temperature transients. Sodium-logging of at least 1% of the fuel rods should be assumed. Approximately 1% of the ^{14}N in nitride fuels will be transmuted to ^{14}C, and another 0.5% to ^{11}B, assuming fuel burnups of 100,000 Mwd/metric ton.

Cladding. - The significant effects of irradiation on the cladding are reduced ductility and swelling. Fully irradiated stainless steel at temperatures of about 1200°F exhibits significantly reduced bonding strength at the grain boundaries. At lower temperatures, below about 1000°F, the stainless steel will apparently regain ductility approximating that of aluminum lawn chair tubing. It is anticipated that irradiated elements, which will have relatively cool outer rods, will withstand a moderate amount of bending and flexure without fracture.

3.3 Aqueous Processing Technology

3.3.1 The Nuclear Fuel Services (NFS) Processing Plant

The NFS plant at West Valley, New York, serves as a good example of present-day aqueous processing technology. This plant[22] is a multipurpose facility capable of handling all LWR fuels that are currently being used. The process employed at NFS consists of a shear-leach mechanical head-end treatment followed by several cycles of tributyl phosphate (Purex) solvent extraction and ion exchange (see Fig. 3.1).

The fuel is received in shielded, water-cooled casks, which may be brought in either by truck or by rail. The casks are deposited in a 44-ft-deep cask unloading pool by means of a 100-ton crane. The fuel elements are transferred to storage baskets and, in turn, these baskets are transferred to a storage pool by smaller cranes. The storage racks are spaced to ensure that they are critically safe.

Fuels are cooled for at least 150 days prior to processing. After this cooling period, the fuel is transferred to the first mechanical

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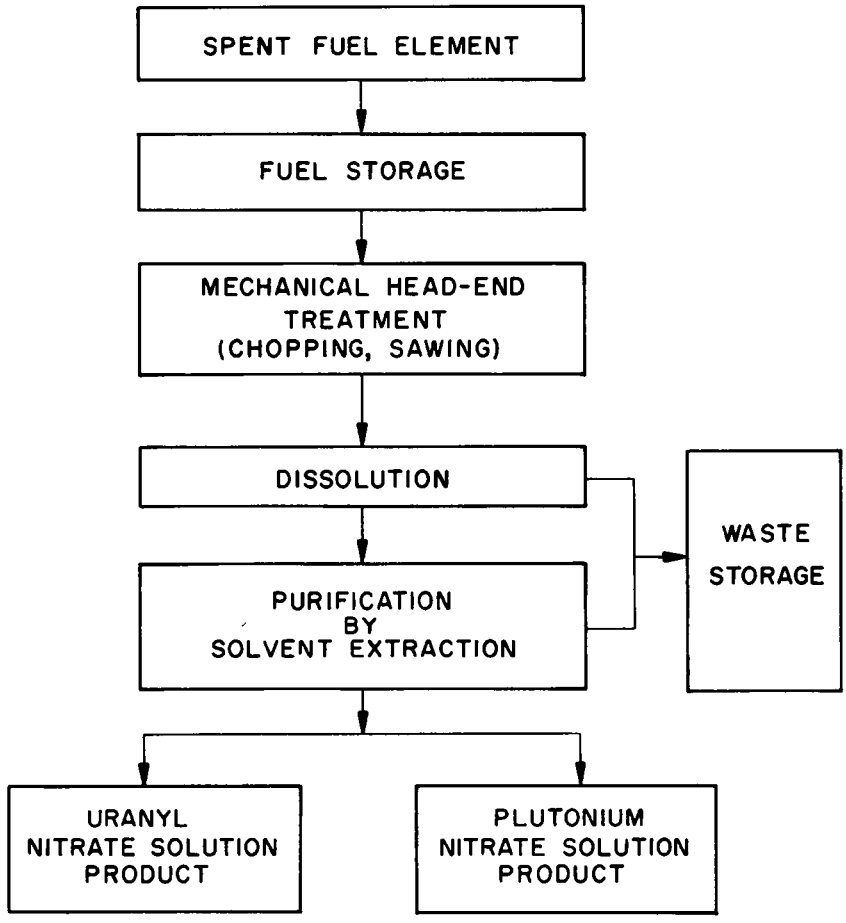

Fig. 3.1. Outline of Process for Processing Spent Fuel.

cell by cranes and a conveyor. In this cell, the end portion of each fuel element is cut off by a remotely operated saw and, if necessary, castings are removed. The prepared element is then loaded into the shear magazine, where it is pushed, by a ram, into the shear. The shear is a 300-ton hydraulic unit in which the fuel can be sheared into pieces ranging in length from 1/2 to 2 in., as desired. The sheared pieces fall into a chute and are collected in chopped fuel canisters.

The canisters are perforated boron--stainless steel cylinders 8 in. ID by 7 ft high. Because shearing disintegrates much of the fuel into powder, the canisters are fitted with consumable liners made of 10-mil mild steel. When full, the canisters are transferred into the first of the chemical processing cells in which dissolution is accomplished.

Description of the Process. - Essentially complete dissolution of oxide fuels can be achieved in about 2 to 4 hr. Metal fuels require about one day. Treatment of the off-gas includes passage through a downdraft condenser on the dissolver, a secondary condenser, a scrubber, an iodine removal apparatus with silver reactors, and, finally, filtration through high-efficiency particulate air filters. The filtered discharge is exhausted through a 63-m stack.

The fuel solution is transferred to a feed adjustment and accountability tank. The leached hulls are rinsed with water, and this rinse liquid is added to the fuel solution. After the hulls have dried, the baskets are dumped for inspection and sampling to determine what losses may have been experienced. The hulls are then sealed in 30-gal drums, transported from the plant in a shielded trailer, and buried as solid waste. Since the radiation level of these hulls is as high as 10,000 r/hr, this operation must be performed remotely.

The volume of the fuel solution in the accountability tank is measured accurately, and multiple samples of the solution are withdrawn for analysis. This analysis represents the feed input to the system and establishes the financial accountability of the plant. It is also important from the standpoint of criticality control for the plant since

limitation of the fissionable material concentration is the primary safeguard.

The recovery and decontamination of uranium and plutonium from the fuel solution are accomplished via a Purex solvent extraction process in which the solvent is tributyl phosphate (TBP) dissolved in n-dodecane (see Fig. 3.2). In the first solvent extraction cycle, the nitric acid solution containing the uranium, plutonium, fission products, and various impurities is contacted with the solvent in a pulsed column. The tributyl phosphate complexes preferentially with uranium and plutonium (or thorium), thereby making them soluble in the organic phase. This leaves almost all the fission products in the aqueous phase, which subsequently becomes the high-level radioactive aqueous waste from the process.

The organic phase, which contains the uranium and the plutonium (or thorium), is scrubbed with nitric acid to remove additional impurities and is then transferred to a second pulsed column. In this second column, the uranium is separated from the plutonium by contacting the organic phase with a nitric acid solution that contains ferrous sulfamate, a reducing agent. The plutonium, which has been tetravalent up to this point in the process, is now reduced to the less-extractable trivalent state so that it can be removed from the column in the aqueous phase. The uranium is then stripped from the organic phase with dilute nitric acid in a third column.

Two additional cycles of solvent extraction are used to complete the purification of the uranium product. The plutonium product is processed through one additional solvent extraction cycle, followed by anion exchange treatment to achieve final purification and concentration; it is then further concentrated in a titanium evaporator and transferred to bottles for shipment.

The acid waste streams are evaporated: (1) to concentrate them for storage, (2) to decontaminate the excess liquid, and (3) to recover nitric acid for recycle (see Fig. 3.3). The high-level radioactive waste from the first extraction cycle (HAW) is evaporated, and the concentrate is sent to the high-level radioactive waste storage tanks. The condensate is

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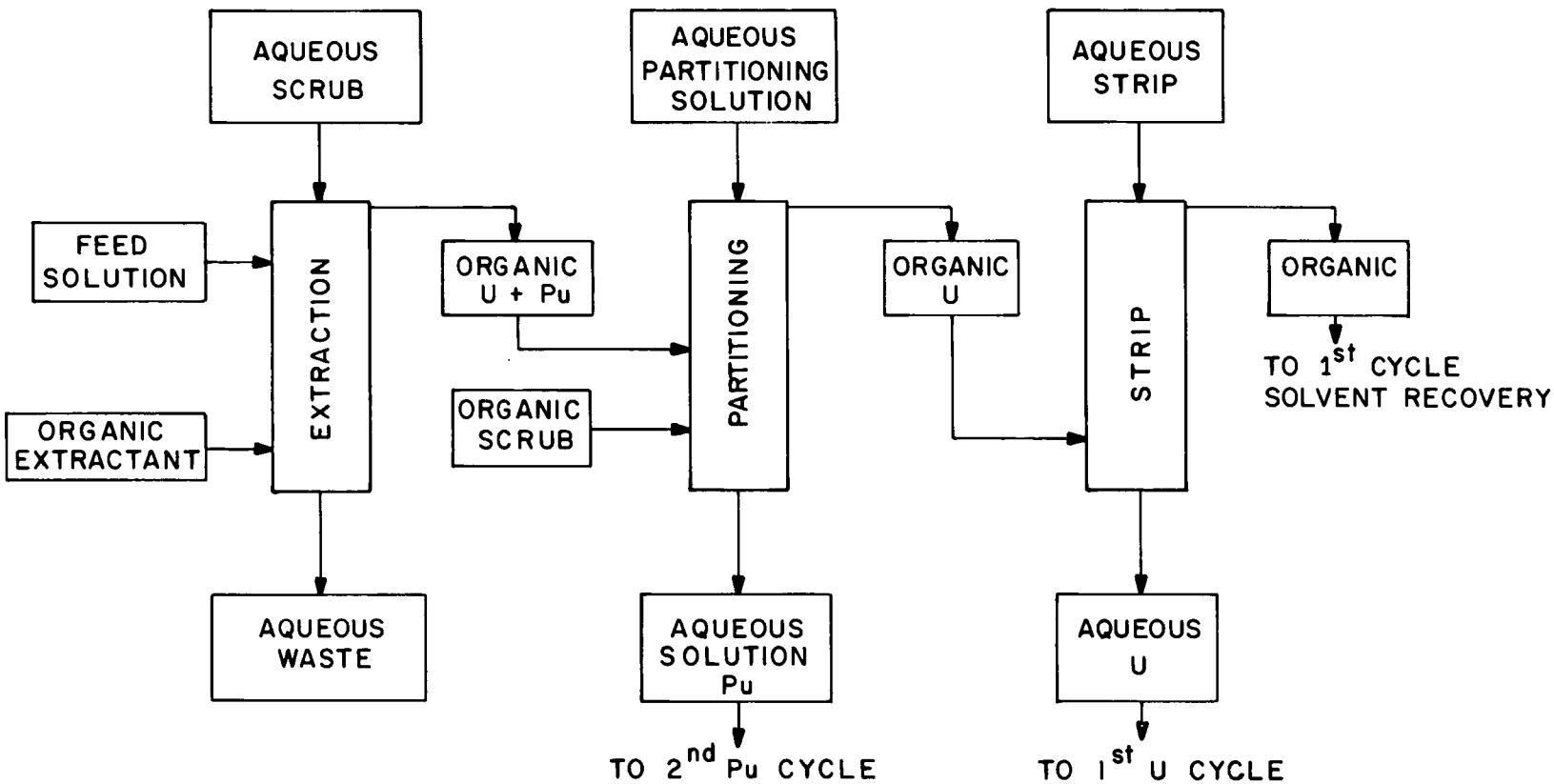

Fig. 3.2. Partitioning Cycle.

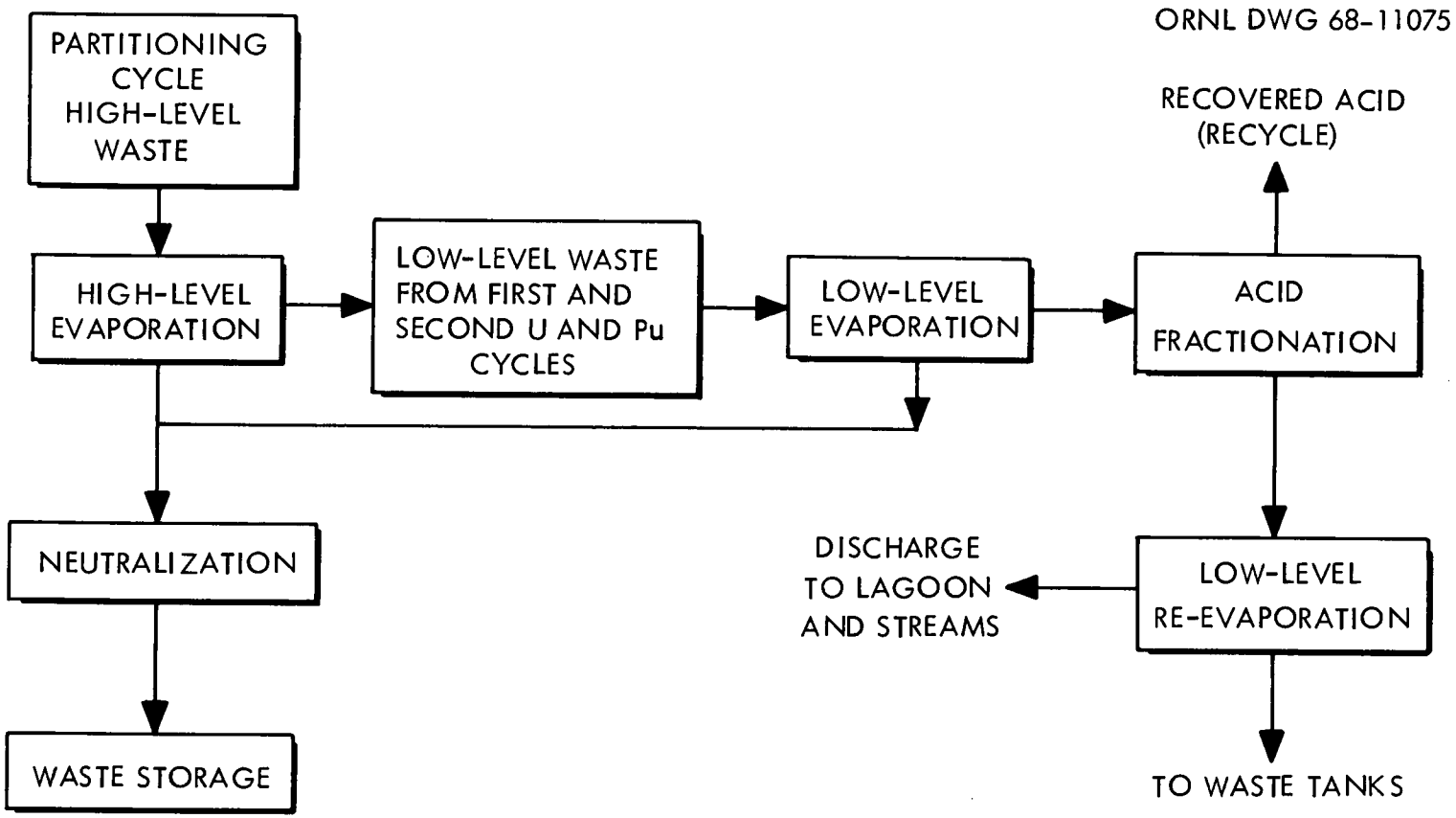

Fig. 3.3. Waste Treatment at Nuclear Fuel Services, Inc.

combined with low-level waste from other processing cycles and evaporated a second time. The resulting concentrate is combined with the first-cycle concentrate. The overhead vapors pass into a fractionation column for recovering the nitric acid, and are then condensed and re-evaporated before the condensate is discharged to the lagoon and on-site streams. Thus in effect, the liquid from the high-level waste is purified by four distillations prior to discharge.

The high-level radioactive waste concentrate is neutralized and stored in 750,000-gal waste tanks that are buried underground. These wastes will be expected to boil for several years. The overheads are condensed and either returned to the storage tank or re-evaporated and discharged via the interceptor-lagoon system.

Operating Experience.[23] - The NFS plant has processed a total of 420 tons of irradiated fuel - about 300 tons of zirconium-clad uranium-metal fuel from the NPR, 50 tons from the Yankee Reactor, 500 tons from the Dresden Reactor, and 16 tons of thorium fuel from Con Ed. The latter three are oxide fuels.

In general, the performance of the plant has been very good. The overall dollar-value recovery has been 98.45%. The uranium and plutonium losses have averaged less than 1% and about 1.5%, respectively. The ^{233}U loss from the Con Ed fuel was only 0.2%. The most radioactive material that has been processed to date was fuel that had been irradiated to a burnup of 18,000 Mwd/ton and cooled 18 months. Satisfactory fission product decontamination factors were obtained.

The performance of the shear has also been good. Although some maintenance and repair were necessary, both were accomplished remotely without excessive downtime.

The existence of some design deficiencies in the plant, mainly in the acid recovery system and the ventilation system, has become evident. Two filters in the off-gas system have failed, releasing some activity to the environs. However, in each case, the release was considerably below the maximum permissible level. In addition, the routing of certain

off-gas lines has created radiation fields in certain work areas. Corrective measures are being applied to solve these problems.

Modifications Necessary for LMFBR Fuel Processing. - The length of the proposed LMFBR fuels and the fact that many rods may be sodium-logged make the present receiving, handling, and mechanical head-end equipment at the NFS plant inadequate for routinely processing LMFBR fuels unless the fuel is dismantled and very carefully cleaned and inspected at the reactor station before shipment. The underwater receiving and storage facilities at NFS would present a serious hazard for fuels that contained sodium-logged pins. The reaction of sodium could also be a problem after chopping the fuel either in the cell or in the dissolver.

The mechanical head-end facilities cannot handle fuel with a length greater than about 12 ft. This limitation would require that the end hardware be cut from the elements at the reactor site. A dismantling facility to accomplish this at each reactor would be prohibitively expensive. This may not be true for 10,000-Mw (electrical), or larger, stations in the future.

For the reasons discussed above, it would be necessary to add new receiving, storage, and mechanical head-end facilities at the NFS plant if LMFBR fuels were to be routinely processed there.

No significant changes would be required in the dissolution, feed adjustment, solvent extraction, and waste disposal facilities for LMFBR fuel cooled 180 days. Flowsheet modifications, such as dilution and use of soluble poisons, could accommodate the higher concentrations of plutonium and fission products in the LMFBR fuel.

The present plutonium isolation and packaging facilities would be inadequate, both in capacity and shielding. NFS is in the process of upgrading the present plutonium isolation system and adding shielding, but it is doubtful that the capacity will be increased to the 50 kg/day necessary for routinely processing 0.5 ton of LMFBR fuel and blanket per day.

3.3.2 Other U. S. Processing Plants Now Planned

The Midwest Fuel Recovery Plant. - The General Electric Company is now building a processing plant,[24] known as the Midwest Fuel Recovery Plant, at Morris, Illinois. The design of this plant is based on the proprietary Aquafluor* process and is specifically tailored for the recovery of plutonium, neptunium, and uranium from fuels that have been irradiated in light-water-moderated and light-water-cooled central station power reactors. The plant is designed to handle, annually, 300 metric tons of uranium in the form of compacted UO_2, initially enriched to no more than 5% ^{235}U and clad in stainless steel or zirconium alloy. The design is based on fuel irradiation at specific powers of up to 30 kw per kilogram of uranium, total exposures of up to 40,000 Mwd/metric ton, and cooling times (from reactor shutdown to initial processing) of about 160 days.

An underwater transfer device is provided for moving the fuel assemblies, after cooling, from the storage basin to the mechanical cell for processing (see Fig. 3.4 for the process flowsheet). In this cell, the fuel assemblies are removed from storage baskets and placed on a disassembly table, where the external hardware is removed by appropriate means and the rods are separated. The individual rods are then fed into a shear where, at a controlled rate, they are chopped into 3-in. segments that fall directly into the leacher. The fuel leacher is a totally enclosed, horizontal-slab, vibrated-tray design of favorable geometry. Over an approximate 20-hr period, one metric ton of sheared fuel is fed to the leacher and contacted with the quantity of recirculating hot nitric acid that is sufficient to completely dissolve the UO_2. The cladding hulls are washed, monitored, and conveyed to a water-filled storage vault.

A single solvent extraction cycle of the Purex type is used for the initial separation of uranium, plutonium, and neptunium from fission products. Pulsed columns are used for extracting and stripping the uranium-plutonium-neptunium product. This product is concentrated for the next step in the process, and the solvent is recycled after treatment.

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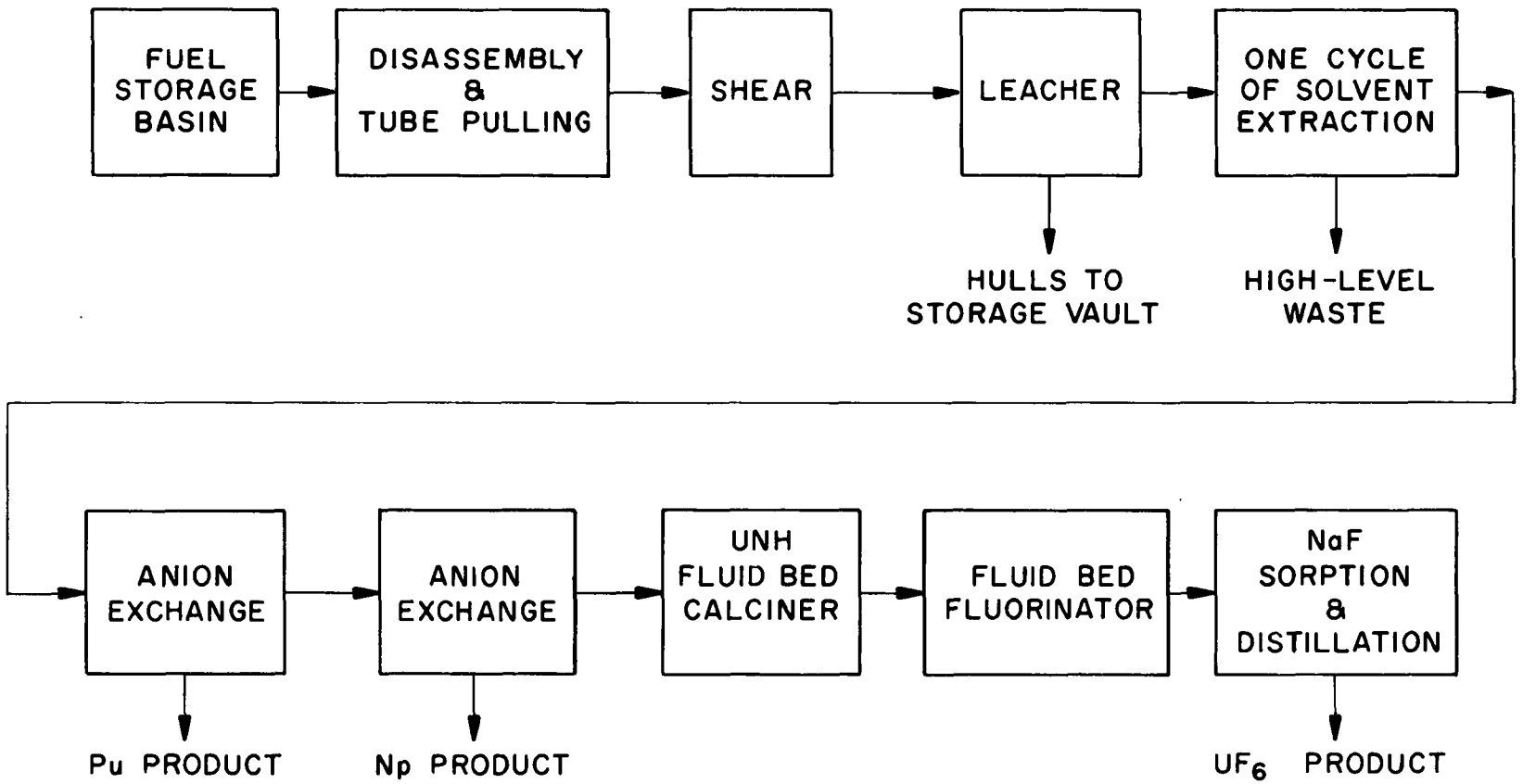

Fig. 3.4. Schematic Process Flowsheet for MFRP.

Two semicontinuous anion exchange contactors, in series, serve to partition the plutonium, neptunium, and uranium into separate product streams and to provide additional fission product decontamination for the plutonium and neptunium.

The purified plutonium product stream from the anion exchange system is transferred to an electrically heated evaporator for concentration to levels suitable for shipping.

The uranium-bearing stream from the anion exchange step flows to an evaporator, where it is concentrated to levels that are appropriate for subsequent calcination to UO_3. The concentrated uranyl nitrate solution is fed to an electrically heated, fluidized-bed calciner that is similar to those used in AEC feed material preparation facilities. Uranium trioxide product is removed from the bottom of the calciner and transferred continuously, by means of a screw conveyor, to a fluidized-bed fluorinator. Here it is mixed with fused alumina particles and contacted with elemental fluorine in a fluidizing gas to produce UF_6.

The hot fluorinator off-gas is passed, first, through a cyclone separator to remove particulate contaminants and, then, through a sorption bed of sodium fluoride pellets to remove trace amounts of the volatile fluorides of plutonium, niobium, and ruthenium. It is then cooled and passed through sintered metal filters to a cold trap system for separation of the UF_6 from the fluidizing gas.

A single-step distillation process serves to separate the UF_6 product from final traces of high-boiling fission products.

The Allied Chemical Barnwell Nuclear Fuel Plant. - The proposed Barnwell Nuclear Fuel Plant[25] (BNFP) of the Allied Chemical Corporation will be constructed on a site of approximately 1700 acres, currently a part of the Savannah River Plant Site, about 5 miles west of Barnwell, South Carolina. The BNFP will be a single-purpose plant; that is, it will process only irradiated fuel elements from light-water central nuclear power stations. However, it will process both slightly enriched uranium oxide and recycle plutonium oxide fuel. Its nominal daily throughput will be 5 metric tons of uranium that has been irradiated to a burnup of

up to 40,000 Mwd/metric ton, at a specific power of up to 50 Mw/metric ton, and allowed to decay for 160 days. Plant products will consist of uranium and plutonium, and possibly neptunium, in the form of nitrate solution.

The Allied Chemical plant will utilize a chop-leach head-end treatment, with semicontinuous dissolution of the oxide core. An adaptation of the Purex process which uses 30% tributyl phosphate (TBP) in a hydrocarbon diluent as the organic extractant, will be the primary basis for separation and purification of products. Extraction will be supplemented in certain cases by ion exchange and special absorption methods.

Irradiated fuel elements will be received at the plant in shielded casks transported by rail or truck; they will be removed from the casks and stored under water. The elements will be chopped into selected lengths, and the chopped segments will be leached by nitric acid. Adjusted dissolver solution will be extracted by TBP. Neptunium would be extracted from the high-level radioactive aqueous raffinate remaining from the initial extraction. It would be further purified by a cycle of ion exchange and then given a final treatment with silica gel. The organic containing the uranium and plutonium from the initial extraction step will be partitioned to separate the plutonium from uranium. Two stages of extraction will complete the plutonium purification system. The uranium will be subjected to an additional extraction cycle, and final traces of radioactive zirconium will be removed by silica gel. Iodine removal, nitric acid recovery, and solvent cleanup systems will be included.

The following relatively new design features will be included in the proposed BNFP:

(1) The fuel will be moved and stored in portable multiple-fuel element storage canisters.

(2) Fuel elements will be chopped in the shear without prior removal of end fittings or segmentation of the elements.

(3) The chopped segments of fuel elements, including end portions, will be charged directly to the dissolver while dissolution is under way in order to regulate the release of off-gas as well as to control the dissolution rate.

(4) A soluble nuclear poison will be added to the nitric acid dissolvent to maintain nuclear safety in the dissolver.

(5) The centrifugal contactors developed at the Savannah River Plant will be used for the first-cycle codecontamination and the partition of uranium and plutonium.

(6) Uranium(IV), with hydrazine added as a holding agent, will be used as the reductant for the partitioning of plutonium from uranium and for aqueous stripping in plutonium extraction cycles in order to reduce radioactive waste volume.

The Atlantic Richfield Company. - The Atlantic Richfield Company plans to build a processing plant,[26] known as the ARCO plant, that will be capable of processing 5 tons of irradiated fuel per day. Initial operation is scheduled for 1974. Options are being obtained on a 2500-acre site near Leeds, South Carolina.

The ARCO plant will employ a shear-leach head-end treatment that is similar to the basic ORNL design. However, the treatment at ARCO will reflect the latest fuel element designs and will be optimized for large-scale operation. Recovery and decontamination of the fuel solution will be accomplished by means of solvent extraction, using a Purex flowsheet modified to allow the recovery of neptunium.

Plutonium and neptunium products from the plant will be in the form of nitrate solutions. Uranium product will be returned to the customer either as uranyl nitrate or as UF_6, as desired.

The plant will accept irradiated fuel from casks transported by either truck or rail. The unloading facility will be optimized for the two basic fuel element designs (BWR and PWR) to permit the handling of the large number of shipments required for a plant of this size. The company also plans to offer a shipping service for irradiated fuel.

3.3.3 Major Foreign Processing Plants

The major foreign processing plants, along with their capacities and the types of head-end treatment employed, are listed in Table 3.6.

Table 3.6. Major Foreign Nuclear Fuel Processing Plants

Name	Location	Capacity[a] (tons/year)	Type of Head-End Treatment	Process Used
Eurochemic	Mol, Belgium	100	Chemical declad[b]	Purex
Trombay	Trombay, India	100	Chemical declad	Purex
Marcoule	Marcoule, France		Mechanical	Purex
La Hague	La Hague, France		Mechanical	Purex
Windscale	Sellafield, England	2500	Mechanical	Purex

[a] Based on low-enrichment uranium.

[b] Eurochemic plans to convert to a mechanical head-end because of the large volume of chemical declad wastes now being produced.

All of these plants use the basic Purex process. The smaller plants, which are really pilot plants, have not been included. The two small Italian plants, PCUT and EUREX, use aqueous technology. Aqueous technology will also be used by the pilot plant planned at Karlsruhe.

The Italians, the Japanese, and the Indians have announced plans for building large processing plants. The Japanese plan a 200-ton/year plant at Tokai-Mura. The Italians plan a 1000-ton/year plant at a location as yet unannounced. The Indians are building a 300-ton/year plant at Tarrapur. All of these plants will utilize the basic aqueous technology of a mechanical head-end treatment and the Purex process.

3.3.4 Trends in Processing Technology

The most outstanding trend in nuclear fuel processing plants is to larger size. This is because the estimated unit cost of processing decreases markedly with increasing plant size. Other costs, such as shipping and accountability for individual reactor fuel batches, increase with plant size so that one would expect the population of reactors in a given area to limit plant size. But, based on present estimates, there will be a sufficient number of LWR's operating when the LMFBR's become commercial to support several 5-ton/day, or larger, fuel processing plants in addition to the two 1-ton/day plants we expect to have in the United States. These plants will be single purpose; that is, they will be designed for processing LWR fuels. It is doubtful that these larger plants will contain the necessary modification, such as parallel head-end and larger plutonium isolation equipment, for the early LMFBR fuels. However, it seems reasonable that a 1-ton/day plant might be modified to process at least long-cooled LMFBR fuel until the load reaches the size that would justify a larger plant built specifically for LMFBR fuel.

As the plants get larger, the degree and reliability of containment for radioactive materials must increase. Thus, we can expect smaller off-gas streams, efficient systems for removing iodine, rare gas isolation and storage, and vaporization of tritium instead of disposal in liquid water.

There will be a trend toward more continuous and automated operation in the head-end step and dissolution, and toward parallel or spare pieces of critical equipment. The newer solvent extraction contactors, the Savannah River-developed centrifugal or ORNL stacked-clone contactors, will probably replace pulsed columns, not because of solvent damage but because of their operational simplicity.

Finally, the established trend is to use the mechanical head-end -- aqueous processing technology in the new larger plants. In this respect, it is interesting to note that all of the new processing plants have chosen the standard Purex process. The advantage of the Aquafluor process for LWR fuels, that it produces UF_6 ready for return to a diffusion plant, does not exist for the LMFBR fuels. This does not mean the MFRP could not be modified and used for LMFBR fuel, but rather that the uranium may be stored as slightly contaminated calcined oxide and the fluorination part of the plant would not be used.

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4. THE APPLICATION OF AQUEOUS PROCESSING TO LMFBR FUEL

The Purex solvent extraction process is employed in nearly every major fuel processing facility, both in this country and abroad. Its efficiency and versatility, and the vast operating experience available from major processing facilities, make it the logical choice for adaptation to LMFBR fuel processing.

In this section, each step of the Purex process (Sect. 3.3.1) is assigned a task number and broken down into process options as shown in Figs. 4.1 and 4.2. The special problems posed by LMFBR fuel are assessed, and possible solutions are described. The higher plutonium content of LMFBR fuel (as compared with LWR fuel) is expected to necessitate modifications in the chemistry of dissolution and solvent extraction and in the process equipment design to provide criticality safety; also, the high decay heat is certain to affect fuel transport concepts. The presence of sodium will require a sodium deactivation step that is not required for LWR fuel processing. If short-cooled processing proves desirable to minimize the inventory penalty on the plutonium values of the fuel, the retention of fission product ^{131}I will have to be improved by many orders of magnitude.

The special problems posed by LMFBR fuel are quite severe in some cases; in general, however, all of them appear to be capable of solution by a systematic research and development program.

4.1 Shipping - Task 1

4.1.1 Introduction

Although power reactor spent fuel has been shipped for a number of years, the art of fuel shipment is yet in its infancy. Fuel shipments have generally been relatively small and have generated, in most cases, no more than approximately 20 kw of heat per cask. Under such circumstances it has been reasonable to use relatively simple means for dissipation of decay heat.

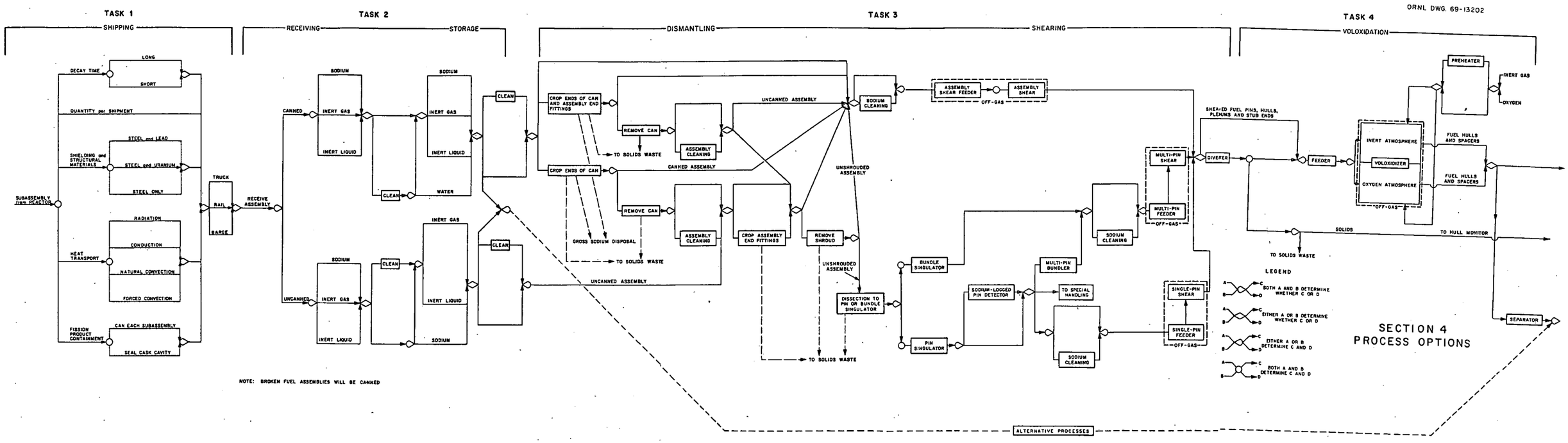


Fig. 4.1. IMFBR Fuel Aqueous Processing. Process Options for Tasks 1-4.

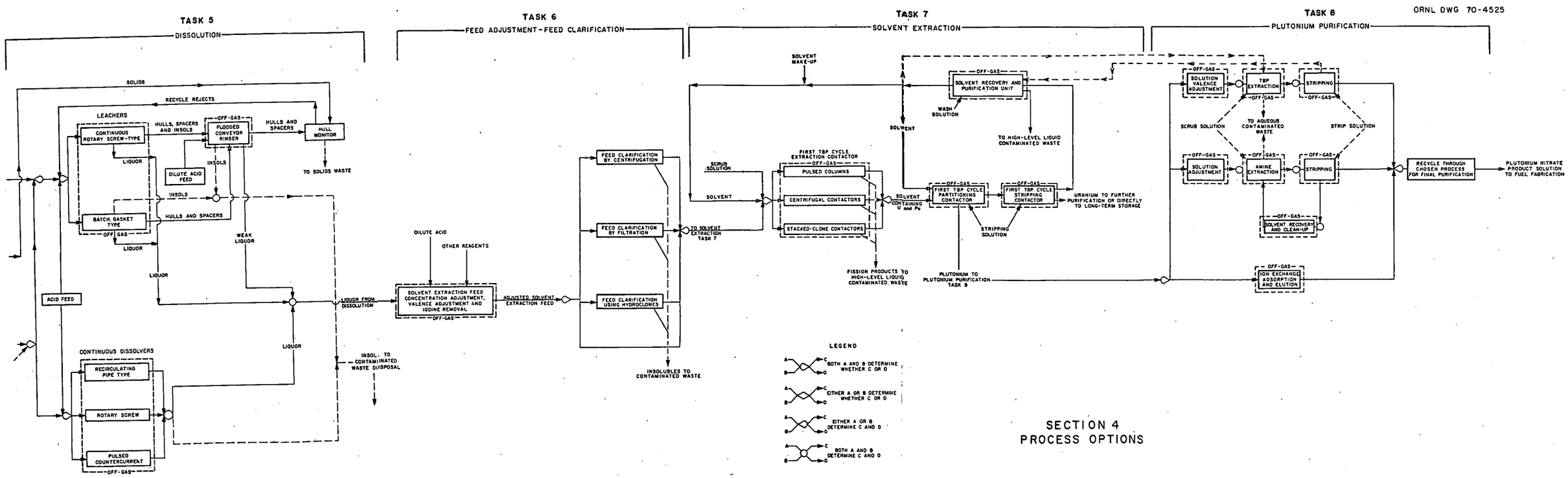
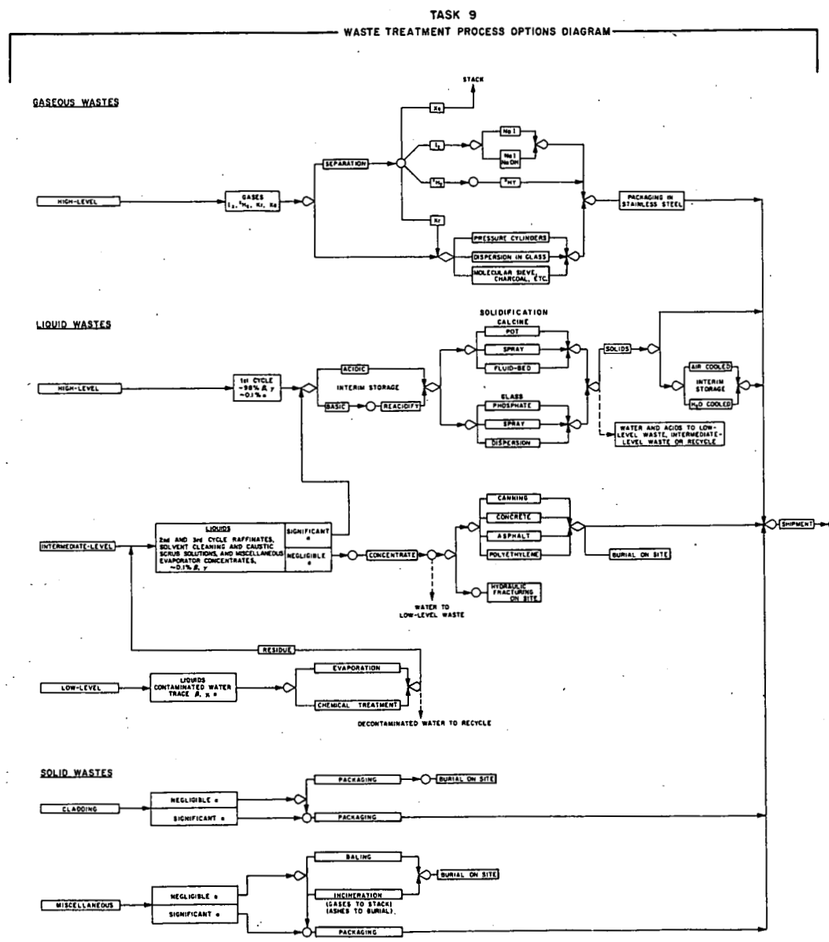


Fig. 4.2(a). LMFBR Fuel Aqueous Processing. Process Options for Tasks 5-8.



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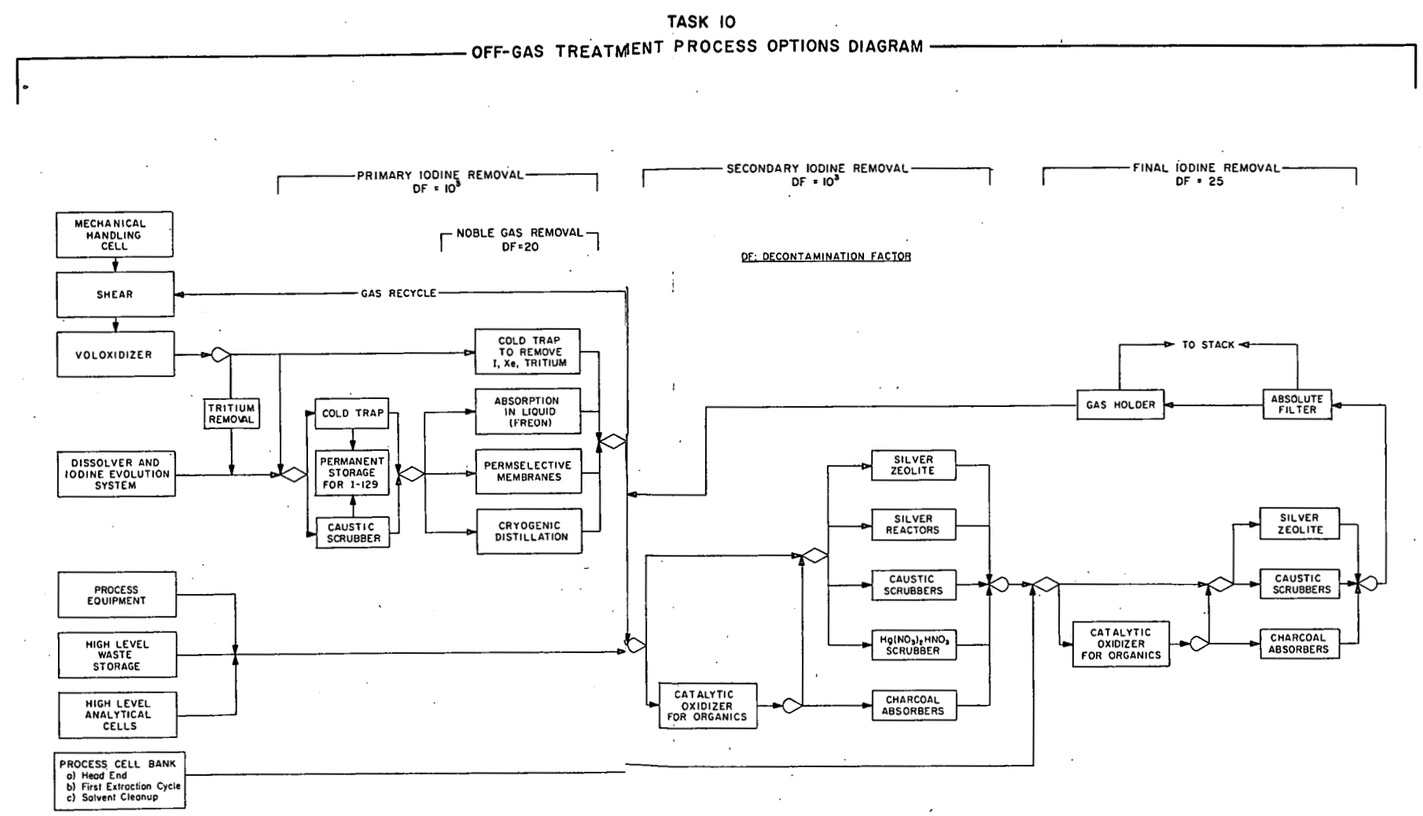

Fig. 4.2(b). IMFBR Fuel Aqueous Processing. Process Options for Tasks 9 and 10.

LMFBR spent fuels will have been irradiated at higher specific power to greater irradiation levels than present-day or currently anticipated LWR spent fuels. Due to the configuration of the fuel and to its high specific power while in the reactor (see Sects. 3.1 and 3.2), we can anticipate greater problems with heat dissipation during shipping (even at the same decay times) than will be experienced with spent LWR fuel.

At this point in time we cannot predict what the optimum decay period or rate of decay heat generation for LMFBR fuels will be. We can be certain, however, that an economic driving force will exist throughout the entire fuel reprocessing and refabrication cycle for minimization of out-of-reactor time and, consequently, shorter decay periods. The heat dissipation problem will obviously be aggravated by this eventuality.

All shipments of spent fuel are subject to applicable Department of Transportation (DOT) regulations. The specific DOT regulations which are expected to have the greatest effect on cask design (and consequently on shipping economy) specify the maximum allowable release of contamination and reduction of shearing under accident conditions as described in 10 CFR 71. (In brief, these regulations state that a cask must be able to survive a 30-ft drop onto a flat unyielding surface, a 4-ft drop onto a 6-in.-diam piston, a 30-min duration 1475°F fire, and an 8-hr immersion in water without excessive release of radioactivity or reduction in shielding.) While criticality must be considered for all shipments containing fissile material, it usually presents no major problem.

For LWR fuel shipments, full reliance has not been placed on the capability to retain the fluid content of a cask under accident conditions. As a result, air has generally been used as the emergency-case primary coolant. For LMFBR fuel shipments, however, it is proposed to use a liquid-metal primary coolant to transfer decay heat from the fuel rods to the cask walls because of its effectiveness in transferring the quantities of heat involved. We believe that the containment of the liquid coolant under accident conditions can be assured through use of a rugged containment vessel that is constructed with improved seals and highly effective energy-absorption devices.

It is realized that this cask design concept represents a rather significant change which will require extensive testing and evaluation to establish its capacity to meet the requirements of 10 CFR 71. However, the achievement of satisfactory LMFBR fuel cycle economics will depend largely upon the development of a capability to ship a practical number of spent fuel subassemblies in a single cask without the necessity for prolonged storage of fuel to reduce its decay heat generation rate, or for predisassembly of the fuel to improve its heat transfer characteristics. The use of liquid-metal primary coolant offers the potential for development of this capability.

It appears likely at this time that interim processing will not be accomplished at as short a decay time as probably will be economic when fuel is processed from a large number of commercial LMFBR's in a plant equipped for economic high-throughput operation. Hence, the shipping casks used to carry fuel subassemblies from demonstration and early commercial reactors may well be different from those which will be used to transport fuel from later commercial reactors to large-scale LMFBR fuel processing centers.

It should be noted that the improvements made in the area of LMFBR fuel shipment will very probably be applicable, in a modified fashion, to LWR fuel shipment.

4.1.2 Cask Requirements and Design Considerations

DOT cask requirements and design considerations are involved principally with release of contamination, maximum dose rate, temperature limits, and contamination of primary coolants.

Release of Radioactive Material. - The maximum amount of radioactive material that can be released is (1) 0.1% of the total radioactivity of the package, or (2) 0.01 curie of group I radionuclides, 0.5 curie of group II radionuclides, 10 curies of groups III and IV radionuclides, except that inert gases are limited to 1000 curies.[1] Release of contamination from the cask is most likely to occur as a result of impact or involvement in a fire.

If a cask containing water coolant is involved in a fire, high pressures will be developed in the fuel cavity owing to the vaporization of the water. Generally, such a cask is designed with a pressure relief valve. Once the valve is actuated, it is most difficult to ensure that it will reseat, and it must be postulated that all water coolant will escape. However, if the radioactive material is sealed within a container in which the coolant has a low vapor pressure (e.g., sodium), and if the system is such that a short-duration fire will not produce excessive temperatures (which is the case for steel-shielded casks normally maintained near ambient temperature), the inner line of containment will almost certainly remain intact and thus allow no contamination to be released to the environment. Although it is not current practice to ship radioactive materials in sealed, sodium-filled casks or in steel casks (i.e., in the United States), the design and construction of such systems are not beyond the capability of current technology.

Increased Dose Rate. - Regulations permit increases in total (neutron plus gamma) radiation level as a result of an accident to a maximum of 1000 mr/hr at a distance of 3 ft from the cask surface. Of the cask types considered, the lead-shielded cask appears to be the most vulnerable, as regards loss of gamma shielding, if it is involved in the accident sequence described in 10 CFR 71. Lead can move under impact conditions; it has a low melting point and a high coefficient of expansion during heat-up and melting which could cause rearrangement of the lead and subsequent loss of shielding from important areas when resolidified. This is not to say that lead-shielded casks cannot be designed to meet regulations; however, the design may be somewhat complex, resulting in increased capital costs. Impact and fire could create excessive stresses in steel and uranium casks, but the potential loss of these shielding materials appears to be less than for lead when exposed to the same environment. Also, loss of the neutron shielding component (which is quite likely if liquid shields are utilized) during an accident would be more severe in the case of a lead-shielded cask than in that of either steel- or uranium-shielded casks owing to the larger thickness of this component required to meet normal dose rate levels.

Temperature Limits. - DOT regulations (49 CFR 173.393) do not place specific temperature limits on the cask shield or on the material being transported. Accessible surface temperatures are limited by regulation to 180°F. Cask surfaces can attain much higher temperatures than 180°F without violating the regulations, provided a barrier is placed over the hot surfaces to make them inaccessible. The maximum practical surface temperature of a lead-shielded cask is about 400°F. Even at 400°F, thermal cycling, thermal shock, and lead expansion during the operating cycle must be accommodated in the cask design.

Both steel- and uranium-shielded casks may be operated at temperatures considerably higher than can lead casks. Considering the fast-reactor fuel heat load, the capacity for elevated temperature operation will likely be an overriding factor in the choice of shielding material for LMFBR fuel casks.

Regulations do not prohibit fuel melting per se. Some of the possible consequences of a fuel meltdown (criticality and release of fission products to the environment) are, however, prohibited. As a practical matter, it is desirable to restrict the temperature of the fuel to levels which are compatible with cladding integrity, thus avoiding gross escape of fission products to the primary coolant even under the hypothetical accident conditions.

Contamination of Primary Coolant. - Primary coolant is defined as the gas, liquid, and/or solid that is in contact with the radioactive material and is used to remove decay heat. Under normal operating conditions, radionuclide concentrations in the primary coolant are limited to 10^{-7} curie/ml for group I, 5×10^{-6} curie/ml for group II, and 3×10^{-4} curie/ml for group III plus group IV. When fuels are shipped in a cask whose coolant will be totally retained and shielded, the requirements pertaining to release of radioactivity will obviously be met and the radioactivity of the coolant will be a matter of concern only to the cask recipient. Therefore, the provisions of this requirement may be waived when the containment vessel is proven to be essentially inviolable.

4.1.3 Current Practice

Cask designs in the United States have, almost exclusively, utilized lead shielding. The United Kingdom prefers steel as a shield material in spite of the weight disadvantage because steel affords additional integrity during a fire. Uranium- and tungsten-shielded casks have been designed and built, but usually for a specific purpose. In this section, current practice in regard to lead-shielded casks will be discussed.

Heat Transfer. - To date, the problems associated with heat transfer have not been great. Heat loads have generally been in the range of 10 to 20 kw per cask and have been distributed fairly uniformly throughout the cavity. Air has been used customarily as the primary coolant because it cannot be "lost" in an accident. In order to accommodate an exceptionally high heat transfer load, metal shot[2] rather than air has been used to keep fuel temperatures at acceptable levels.

Two noteworthy casks designed to remove large heat loads may be mentioned. The first is a 125-ton cask designed by the U.S. Navy to serve, initially, as a dockside cooling and storage point for its submarine cores and, after cooling, as a shipping cask. Forced convection of water, capable of removing about 100 kw of heat, is used to cool the core. The second is a 75-ton Westinghouse cask designed to remove 70 kw of heat from spent Yankee Atomic fuel subassemblies.[3] This cask was designed to contain ten Yankee Atomic fuel subassemblies, each consisting of about 300 fuel rods, 0.340 in. OD and spaced 0.422 in. apart in a square array. At cask design conditions, the fuel rods would be emitting 2.9 w per foot of length (90 days decay). This cask, under normal conditions, uses circulating water both as the primary and as the secondary coolant. In an emergency, all coolant is assumed to be lost, leaving the space between the fuel and the inner cask wall filled with air. Under accident conditions, the heat flux at the cask surface will be about 1800 Btu hr^{-1} ft^{-2} of flat surface, and the lead inner surface temperature and the maximum fuel element temperature will be approximately 619°F and 1400°F, respectively.

Cask Design. - Small casks for transporting small quantities of radioactive material are quite often overdesigned with respect to inner and outer shell thickness. Such casks have been tested and found adequate from an impact and puncture point of view.[4] As casks have become larger and the consequences of accidents more severe, designs have become more complex. Due to lack of theory and experiment relative to the protection of casks during accidents, cask designers have resorted to providing the cask with a crushable impact shield and/or fire shield. For example, in the case of the previously mentioned 75-ton Westinghouse cask, the entire rail car is designed to protect the cask during impact.

Special precautions must be taken in the design and construction of a cask to ensure against leakage under all credible conditions. Recent tests[5] have indicated that fins welded to the cask around the closure area can provide enough energy absorption and dissipative capacity to protect the seal integrity even with the cask operating under pressure. At least two lines of containment are generally provided to maintain control of the fission products in spent fuel shipments. One recent design provided three lines of containment, since the fuel contained a sodium bond which was not to come in contact with water when stored in a pool over an extended period of time.[6] The cask seal was maintained by a single stainless steel-silastic rubber gasket, but each fuel subassembly was carried inside two concentric canisters. Each canister was sealed by welding.

4.1.4 Applicability of Present Practice and Current Technology to LMFBR Fuel Shipment

The primary problem centers around the ability of the cask to dissipate the requisite amount of heat both under normal and accident conditions. The heat flux per foot of fuel rod and the cask external heat flux are currently rather modest. In the case of the previously mentioned Westinghouse cask these fluxes are 2.9 w/ft and 1800 Btu hr^{-1} ft^{-2}, respectively. For this case, the accident equilibrium temperature of the fuel is 1400°F, which is acceptable for heavy-walled, stainless-steel-clad LWR fuel rods. However, for an LMFBR fuel rod irradiated

at an average power of 10 kw/ft (a typical power generation rate) for a period of 1 year, the decay heat generation rate will be approximately 15.5 w/ft at 20 days decay, 8.3 w/ft at 60 days decay, and 4.1 w/ft at 150 days decay. (The effect of ±50% changes in irradiation time would be of the order of ±10%.) Further complicating the problem of decay heat dissipation by current techniques is the low ratio of core length to total length of subassembly to be transported. Current practices are not suited for effective axial transport of heat.

As further demonstration of the ineffectiveness of current practice (use of uncirculated air for primary coolant) for dissipation of heat, Fig. 4.3 (see Sect. 4.3) shows the approximate maximum fuel rod temperature within a 217-rod bundle for various conditions of heat generation rate and emissivity. Even with an external temperature of only 212°F (an arbitrary and unrealistically low temperature assumed for the surroundings of the subassembly wrapper), it can be seen that the temperature of the hottest rod is greater than 1300°F (significantly higher temperatures reduce cladding strength to unacceptable values) for any decay heat generation rate greater than 2.8 w/ft when the emissivity is less than or equal to 0.5. The decay heat rate of 2.8 w/ft corresponds to the decay heat generation of a fuel rod irradiated at an average power of 8.5 kw/ft for a period of 1.5 years and decayed for 250 days.

It is, therefore, apparent that an improvement in heat dissipation is desirable for transport of LMFBR fuels (and possibly also of benefit for shipment of LWR fuels). Major improvements in heat dissipation capability can be made through use of an effective primary coolant and of increased external cask temperatures. Sodium is a very effective primary coolant as a result of its good thermal conductivity ($\sim$ 37 Btu hr^{-1} ft^{-1} $°F^{-1}$) and its capacity to transport heat over long distances with relatively low-temperature differentials by means of thermal convection. Much higher external temperatures can be accepted (under the accident condition), provided steel is used for shielding in lieu of lead.

The major problem concerning the use of any fluid primary coolant other than air is retention of the coolant under all conditions. Model

casks of various descriptions have shown satisfactory capability to contain fluids before and after a 30-ft free-fall impact test. The number of tests is very small; they were made with nontypical systems and certainly do not provide the degree of assurance necessary to permit acceptance of a shipping system that is totally dependent on retention of the coolant for safety.

Steel can be made which will have satisfactory performance for cask service at elevated temperatures. The metal thicknesses required for a steel-shielded cask range up to 21 in. Fabrication of such a system is certainly not routine, although it appears to be a problem which can be solved by normal engineering and developmental effort. This developmental effort will include tests to determine the proper material composition and heat treatment. Additional information on this general subject can be found in ORNL-TM-2723, An Engineering Evaluation of LMFBR Fuel Shipment.

4.1.5 Cost of Shipping

The major items to be considered in calculating the cost of fuel shipment are as follows:

1. Fuel handling at both the reactor site and at the processing facility.
2. Cost of cask usage (includes return on investment).
3. Freight cost.
4. Insurance cost.

The cost of fuel handling should include the price of preparation of the fuel for shipment, assuming that these preparation operations do not reduce the cost of fuel recovery at the processing site. An ORNL study[7] has been made which indicated that the cost of preparation will be very significant for a single 1000-Mw(e) reactor station if the preparation operations include complete disassembly of the fuel subassembly and canning of the fuel rods in steel pipes. As the size of the reactor station increases, the unit cost of such fuel preparation will decrease markedly. Reduction of fuel preparation charges to a negligible amount

appears reasonable if there is no disassembly required for shipment, if there is no cleaning required for shipment, and if the same handling equipment can be used for loading fuel subassemblies into the shipping cask as is used for other reactor fuel handling operations. The unit cost of fuel handling at the processing site can be expected to be of minimal importance economically, provided that the capacity of the facility is of the order of tons per day. The item causing fuel disassembly at the reactor site to be of an unacceptable magnitude is the large investment required, coupled with usage for only a small fraction of the time; by contrast, such equipment at a processing facility can be expected to be utilized almost continuously, thus reducing the unit cost for use of equipment.

The expense for cask usage can be expected to be one of the largest costs, if not the largest cost, involved in the shipment of LMFBR fuel. The amount to be charged for this is dependent upon the degree of utilization that can be achieved and, of course, upon the cost of the cask per kilogram of fuel shipped. This cost will undoubtedly be a function of the cask-weight-to-fuel-weight ratio as well as of the complexity of the cask.

The cost of freight will be controlled largely by the weight of cask per weight of fuel, plus the distance shipped. The method of shipment will also have some influence.

The cost of insurance, while not negligible, is not particularly significant. It has been estimated that the insurance cost will amount to approximately 0.05% of the value of the shipment.

If the decay period prior to shipment is extended beyond the time necessary to deliver the fuel to the processing facility at the earliest time that the processing can be economically accomplished, the cost of shipment must include the inventory cost for the incremental period. An appreciation for the magnitude of the inventory cost can be gained by observing that the weekly inventory cost for a material containing 15% fissile plutonium, valued at $10/g, at an interest charge of 10% per annum will be sufficient to pay the freight on 36 times the weight

of the fuel over a distance of 2000 miles. Inasmuch as the ratio of the shipping cask weight to the fuel weight is not strongly influenced by the decay time, but inventory penalty is linear with decay time, it can be seen that it will be economically desirable to ship spent fuel as soon as it can be economically processed.

The effectiveness of heat distribution from the fuel to the cask shielding is the major factor controlling the amount of fuel that can be shipped in the cask. The figure of merit is the ratio of cask weight to the weight of fuel shipped. Table 4.1 gives a comparison of three cask designs, two of which employ the conventional means of heat distribution and one that employs the more effective liquid metal coolant method. The first of these casks is for PWR fuel shipment, and was designed according to current practice; the second cask is for fast-breeder-reactor (FBR) fuel and is designed along the same lines. These two casks employ air as the primary coolant in the emergency case; therefore, heat is rejected from the fuel rods principally by radiation from rod to adjacent rod and thence to colder shielding wall. The last column presents the expected performance for a conceptually designed cask employing sodium as primary coolant. This cask design provides for the shipment of about 1.8 tons of highly irradiated core fuel (UO_2-PuO_2) incorporated in 36 Atomics International Reference Oxide Reactor fuel subassemblies. Thirty-six subassemblies were chosen as a logical load since 36 hexagonal shapes can be situated in three concentric rings in a hexagonal array. Further, the estimated rod and shielding temperatures (when operating under emergency conditions) are well within the acceptable ranges. In this case, it has not been necessary to disassemble the fuel subassembly into rods, as was necessary for the two casks designed to handle relatively high-activity FBR fuels. In spite of the fact that the length of the core section being shipped in the "advanced design" cask is only about 25% of the total length of the cask, this cask has the most advantageous ratio of cask weight to fuel weight of any of those intended for shipment of fuel having decay heat generation rates of 13 w/ft or greater. It can also be observed that the ratio of cask weight to electrical power

Table 4.1. Comparison of Cask Designs

	Current Practice		Advanced Design
Cask design report	WCAP-1859	[a]	ORNL-TM-2723
Reactor	Yankee	GEAP-4418	AI Ref. Oxide
Core fuel per shipment, kg	2740	397	1796
Core length, in.	91.8	16	48
Total length subassembly, in.	111	56	212
Decay heat, w per ft of pin	2.9	14.4	13.5
Decay heat, total kw	70	83.7	422
Max. fuel rod temperature, °F	1400	1227	1155
Cask exterior temperature, °F	446	321	710
Primary coolant	Air	Air + Cu grid	Sodium
Shielding material	Lead	Lead	Steel
Bundle disassembly	No	Yes	No
Cask-to-core ratio, kg/kg	25	192	83
Cask-weight-to-fuel-energy ratio, kg/Mwd(e)	2.08	4.8	2.59

[a]Personal communication from R. Salmon, ORNL, January 1967.

generated is approximately the same for the advanced-design sodium-filled cask as for the cask designed to carry PWR fuel.

Criticality is not expected to contribute materially to either the difficulty or cost of shipment. Only in cases of large fuel shipments (~30 subassemblies or more per cask) will the effective neutron multiplication factor (k_{eff}) exceed unity, and then only for the more reactive types of LMFBR fuel (e.g., AI Ref. Oxide) under optimum conditions. Even in large fuel shipments the degree of supercriticality is such that utilization of nominal amounts of fixed neutron poisons within the cask can provide nuclear criticality safety. Aside from prevention of criticality per se, it may be desirable to include fixed neutron poisons in the cask to mitigate the problem of neutron shielding. That is, subcritical neutron multiplication resulting from interaction of fixed-source neutrons (originating from transuranium elements in the fuel) with fuel nuclei could be reduced through the addition of nonfissionable neutron absorbers.

The fuel assembly configuration will have some bearing upon the cost of shipping as it affects the difficulty of removal of heat from the fuel rods and its distribution within the cask cavity. Without question, higher ratios of the fuel subassembly total length to core length will adversely affect the ratio of cask weight to fuel weight; however, the penalty to be suffered as a result of increasing ratios of subassembly total length to core length is not as large as might be expected. In ORNL-TM-2723, comparisons were made of the weight of a shipping cask for full-length fuel subassemblies with the weight of a cask for cropped subassemblies (cropped to reduce the overall length by more than a factor of 2). In the case of a cask containing 18 subassemblies, the reduction in cask weight attained by reducing the length of the subassembly from 17 ft 8 in. to the cropped length of 7 ft 8 in. was only 12.5%.

4.2 Receiving and Storage - Task 2

4.2.1 Requirements and Objectives

The processing plant facilities for receiving and storage should provide the means for rapid, effective, economical, and safe operation. The specific requirements for this subtask are contingent upon determinations that are made in Tasks 1 and 3; hence, the major part of this work should be delayed until reasonable definition of the nature of the shipping casks and the head-end processing system becomes available. Obviously, however, a limited amount of early work on receiving and storage is required to serve as input to the other two tasks.

The fuel receiving and storage problem will require close scrutiny for the short-term (FFTF and demonstration reactors), as well as for the long-term, need. Minimization of capital outlay for the short-term need should be a major objective but one which will require weighing against other objectives, such as utility over the entire period that LMFBR fuel will constitute a processing load for the plant and the desirability of successful demonstration of the essentials of the system to be used with the larger amounts of fuels to be generated by later commercial LMFBR's.

4.2.2 Present Practice for LWR Fuels and Sodium-Cooled Reactor Fuels

As is well known to all who have even a passing acquaintance with present-day chemical processing plants, the practice is now, and has been in the past, to unload and store the incoming fuel under water. Upon receipt of a shipment, the cask primary coolant is usually sampled for determination of radioactivity level. The cask is subsequently lowered into a water-filled canal, unloaded, and then removed from the canal.

After removal of the cask, it is necessary to check the cask exterior surface for radioactive contamination. Some difficulty is usually encountered in reducing the surface contamination to acceptable levels. Experience at NFS has been that at least two days are required

for the total operation of cask receipt, unloading, and surface decontamination. This operation, as presently carried out, not only precludes productive utilization of an expensive cask for those two days but also consumes a significant amount of manpower.

Because the quantity of fuel generated by sodium-cooled reactors has been small, fuel handling and storage practices for these fuels have generally been similar to those applied to LWR fuels. They have (with one exception) been shipped in the same general type of cask as is used for LWR fuel, and initial operations at the processing site have been similar to those employed with LWR fuels. The exception to this generalization is EBR-II core fuel, which has been processed in an on-site fuel cycle facility. In this situation, the fuel rods are allowed to decay for a short period (about two weeks) in a sodium bath and are then cleaned of sodium in the course of being transferred from the reactor to the adjacent processing facility; subsequently, they are handled in an air-filled cell.

4.2.3 Special Problems with LMFBR Fuels

LMFBR fuels will, by definition, have been immersed in molten sodium during their irradiation, and quite possibly also during transit to the processing facility. There may be sodium inside the fuel rods, either as a result of failure of the rods or by design. The presence, or the threat of the presence, of sodium in the fuel rods will influence the method of handling and storage of LMFBR fuel at the processing plant.

If the fuel subassemblies are shipped in sodium, the receiving facilities should be arranged so that the subassemblies can be withdrawn from their sodium-filled container(s) while materials that react with sodium are excluded. (This shipping method is considered the most probable.) Subsequently, storage of the fuel, with possible prior cleaning to remove external sodium, will be required to provide a surge capacity for feed to the head-end processing facilities. (The various methods of sodium removal are discussed in Sect. 4.3.)

4.2.4 Possible Solutions

A portion of the receiving facilities will, of necessity, be a replica of a portion of the reactor site shipping facilities. A solution that appears to be promising for minimizing turnaround time both at the reactor site and at the processing site is the avoidance of immersion of the cask in a contaminated environment. We believe that the portion of the cask surface to be contaminated can be limited to relatively small areas and that extensive decontamination of these areas will not be necessary since it will be reasonable to cover these areas during transit to prevent spread of the surface contamination. Experimental effort will be required to establish the merit of this approach.

There are a number of possibilities for storage of fuel. These include: (1) storage in water, (2) storage in an inert gas, (3) storage in "inert" liquid metal or other liquid, and (4) storage in sodium. Storage in any of the first three media indicates removal of external sodium. If the uncanned fuel is stored in water, the possibility of the reaction of internal sodium with water exists. We do not know, at this time, precisely what the consequence of this action would be. There are indications that the reaction would be quite mild, producing only a small amount of hydrogen. A major concern in this area could be contamination of the pool water and canal room atmosphere, although this eventuality may not develop with oxide fuels due to their general inertness. (Carbide fuels, on the other hand, would be very reactive with water). Storage of fuel in fluids other than water imposes a variety of mechanical and operating problems, and it is not clear at this time which method is preferable. The choice between the storage methods is, then, one which can be made logically only after a careful evaluation of the various possibilities, and the evaluation probably will require some experimental effort to gain additional information.

4.3 Head-End Processing of LMFBR Fuel - Task 3

4.3.1 Introduction

There are several types of head-end methods by which spent fast breeder reactor fuel might be treated in preparation for the subsequent aqueous Purex recovery operations. These candidate processing methods, along with a summary of appraisal comments, are presented in Table 4.2.

In general, the processing methods reviewed fit into one of the following categories: aqueous decladding (Sulfex[1,2]), molten metal decladding (zinc,[3] copper-antimony[4]), melt decladding,[5,6] simultaneous dissolution of cladding and core (Darex[7] and electrolytic[8,9]), simultaneous cutting of cladding electrically and nitric acid dissolution of core (lixiviation[10]), and the breaking of cladding mechanically followed by core dissolution in nitric acid (shear-leach[11]).

A study of the various processes indicates that the attainment of a safe reaction of sodium is a problem common to all methods. Control of this reaction in the Sulfex, Darex, and electrolytic dissolution processes is much more difficult; therefore, these methods are less likely to be used with LMFBR fuel. In addition, they also dissolve the inert metal cladding, incurring a waste penalty.[12] Sulfex is unsuited for carbide fuels because uranium is precipitated and nitrides are not attacked significantly by the sulfuric acid dissolvent. Presently, there is no information on possible Darex reactions with carbides and nitrides.

The amount of sodium that an electrolytic dissolution process can tolerate from a safety standpoint has not been established. However, it appears to be only a few grams since, in the planned processing of Hallam fuel, the sodium reservoir portion of the fuel bundle is removed, leaving mainly the thin (a few mils) sodium bonding layer. Electrolytic dissolution is also a relatively slow operation, and scale-up of units to a high-capacity plant may not be practical. There are no specific data available on the electrolytic dissolution of carbide and nitride fuels. However, it appears that this method could be used for such fuels.

Table 4.2. Candidate Head-End Methods for Processing Spent LMFBR Fuel

Process	Process Appraisal	LMFBR Usage Prospects
Aqueous Decladding Sulflex: H_2SO_4 Dissolvent	1. Proved at Eurochemic with spent fuel 2. Dissolved cladding incurs waste penalty 3. Loss of plutonium to cladding waste 4. Safe control of sodium reaction difficult 5. Unsuitable for carbides and nitrides	Highly unlikely now or in the future
Molten Metal Decladding A. Zinc B. Copper-Antimony	A. 1. In an advanced stage of development at Argonne National Laboratory, but unproved with spent fuel 2. Separation of cladding melt from core difficult 3. High ratio of molten metal to cladding 4. Reaction with carbides and nitrides unknown B. 1. In early state of development at Euratom 2. Reaction with carbides and nitrides unknown	A. Worthy of continued appraisal B. Too early to appraise, but worthy of continued surveillance and appraisal
Melt Decladding High Temperature Decladding	1. In a very early stage of development 2. Offers chance of eliminating mechanical dismantling and shearing operations plus possible volatile fission gas removal and destruction of sodium 3. Effect on oxide, carbide, and nitride cores unknown	Too early to assess. Worthy of continued development and analysis
Simultaneous Dissolution of Cladding and Core A. Darex: Aqua-Regia Dissolvent B. Electrolytic: HNO_3 + Amperes Dissolvent	A. 1. Developed but unproved with spent fuel 2. Dissolved cladding incurs waste penalty 3. Requires exotic materials of construction 4. Safe control of sodium reaction virtually impossible 5. Reaction with carbides and nitrides unknown B. 1. Proved at Savannah River with spent fuel 2. Safe control of sodium reaction difficult 3. Incurs cladding waste penalty 4. Scale-up difficult 5. Forms organic compounds with carbides	A. Highly unlikely now and in the future B. Good, but needs continued surveillance and analysis
Simultaneous Cutting of Cladding Electrically and Dissolution of Core Electrolytic and Lixiviation: Electrolyte + Amperes + Impinged HNO_3 Dissolvent	1. Early stage of development 2. Unproved with spent fuel 3. Scale-up difficult 4. Safe control of sodium reaction difficult 5. Does not incur cladding waste penalty 6. Mechanical handling problems 7. Forms organic compounds with carbides	Too early to assess. Worthy of continued analysis and appraisal as method develops
Breaking of Cladding Mechanically Followed by Core Dissolution Shearing and Leaching: HNO_3 Dissolvent	1. Proved in commercial reprocessing at NFS, Inc., with spent LWR fuels 2. Does not incur cladding waste penalty 3. Safe control of sodium reaction difficult 4. Forms organic compounds with carbides	Excellent, but needs continued development and appraisal, especially for carbide and nitride fuels and sodium-logged fuel

Molten metal decladding methods involve the handling of large quantities of melt in relation to the quantity of cladding that is dissolved or alloyed. For example, the molten zinc to stainless steel ratio is approximately 15 to 1. These comments apply equally well to the Euratom copper-antimony method[11] now being developed in Belgium, although the ratio of molten metal to cladding in this method is claimed to be much more favorable. No data are presently available for the oxide, carbide, and nitride fuels.

Melt decladding,[5,6] or thermal melting of the cladding, as a backup process for shear-leach appears promising in some very preliminary ORNL tests. As development proceeds, it will continue to be analyzed and evaluated for oxide, carbide, and nitride fuels. Electrolytic cutting--lixiviation,[10] along with the shear-leach process, is now being given serious consideration as a replacement for the Sulfex or Zirflex process at the Eurochemic processing plant in Mol, Belgium. The final choice between the two methods awaits further development of the electrolytic-lixiviation process. The principal advantages claimed for electrolytic-lixiviation in contrast to shear-leach are that: (1) no hazardous particulate matter is produced, (2) longer pieces can be severed and lixiviated than can be sheared and leached (although this is not necessarily true), (3) uranium and plutonium losses are less, (4) it eliminates cropping and dismantling operations, and (5) there is a possibility that spent sodium-bonded fuel can be processed at carefully regulated, safe rates by reducing the temperature of the electrolyte used in the cutting and lixiviation operation. At present, the process is not sufficiently developed to validate or confirm the claims made for it. In its present form, the process does not appear to be one that can be easily scaled up other than by use of a multiplicity of units. The development and progress of electrolytic-lixiviation will be followed and continued to be analyzed as a backup LMFBR head-end method.

Table 4.2 and the analysis just presented indicate that the mechanical shear-leach process is the most fully developed and advanced head-end method for treating LMFBR fuel at the present time. The development of

mechanical processing of spent nuclear fuels began in 1954 with an investigation of the mechanical dejacketing of metallic uranium fuel slugs. This work and many other related mechanical techniques such as disassembly, breaking, sawing, and shearing are described elsewhere.[13] Currently, shear-leach is used as a head-end method for processing LWR fuel at Nuclear Fuel Services, Inc.[14] (NFS), which is the only commercial nuclear fuel processing plant in the United States. It is also being studied by several other companies--the Allied Chemical Corporation,[15] the General Electric Company,[16] the Atlantic Richfield Company,[17] and the National Lead Company[18] --preparatory to possible commercial processing of spent LWR fuel. Foreign applications of the shear-leach process include those of the United Kingdom at Windscale,[19] the West German Republic at Karlsruhe,[20] Italy at Saluggia,[21] France at L'Hague[22] (sodium-bonded Rapsodie fuel) and Fontenay-Aux-Roses,[23] and Japan at a site not yet named.

Using the analysis presented above as a basis, a chart was prepared (see Task 3 in Fig. 4.1) to illustrate the principal head-end processing routes, options, and processing alternatives. The problems brought about by the presence of sodium or other coolants, and their influence on the head-end steps and options, are dealt with separately in a later section (see Sect. 4.3.6 and Fig. 4.8).

As shown in Tasks 2-5 [see Figs. 4.1 and 4.2(a)], canned or uncanned fuel assemblies shipped in sodium, inert gas, or inert liquid coolant are received and surge-stored in water, sodium, inert gas, or inert liquid coolants. The assemblies (canned or uncanned) are withdrawn from storage as needed and sent to the dismantling operation, where the ends of the can (in the case of a canned assembly) and the end hardware of the fuel (in the case of an uncanned assembly), are removed by a cropping operation.

The cropped fuel assemblies, depending on the options shown in the dismantling operation (Fig. 4.1), can be decanned, deshrouded, dissected (reduced to single fuel rods or groups of rods), and then sent to the appropriate shear feeder. Options permit the shearing of fuel as single rods, a multirod array, or as the cropped uncanned or canned fuel assembly. Single-rod and multirod shears may be preferred over bundle

shears because they are small and more cheaply maintained, heat dissipation is simpler, and the sheared product is more readily handled downstream. The final product may be comprised of pieces of fuel rods, wire spacers, grids, shroud sections, and severed reflector rods. The simplest product consists only of severed fuel rods and spacer wires.

The sheared fuel passes through a diverter or valving unit, which routes the empty, unfueled gas plenum sections and the core and blanket portions to separate transfer chutes. A diverter unit does not yet exist; concepts will have to be designed, fabricated, and tested. Sheared pieces containing core and blanket materials are sent to a volatile fission product gas removal step (see Sect. 4.4), and the plenum pieces are routed to an empty hull leacher. In the fission gas release step, sheared fuel can be treated (heated, oxidized, etc.) to remove volatile fission product gases and then be sent directly to a dissolver; alternatively, if it is possible, by practical means, to separate the core and blanket material from the hulls and wire spacers after the volatile fission product gases have been removed, the powder would be sent to a dissolver and the separate hulls and wire spacers would be routed to a hull leacher unit.

Leached hulls, wire spacers, grids, shrouds, etc., are rinsed, monitored, and recycled to the hull leacher (if residual fissile and fertile material is detected) or sent to a step which converts them to a minimal-volume form that is more suitable for handling and waste disposal. All of the processing steps listed will require development and evaluation. The treatment of fuel in the canned or uncanned state merits considerable developmental effort since there is no available reference technology. The disassembly of LMFBR fuel appears to be much more difficult than that of LWR fuel, primarily because of a close fitting shroud. Handling problems, in general, are also made more complex because of decay heat emission and the long lengths (~ 18 ft) involved. For example, if fuel rods or multirod arrays are processed, their frail spaghetti-like structure will require the development of appropriate nondestructive handling techniques.

4.3.2 Decay Heat Dissipation

The decay heat emitted from LMFBR core and axial blanket assemblies presents problems in the head-end handling and processing steps. For example, it restricts the time a fuel assembly, or subassemblies, can be handled without attaining and exceeding a temperature that will seriously weaken it structurally. Heat can also shorten the service life of processing equipment components and cause excessive downtime. Although Strader and Yevick[24] have studied the problem of heat transfer during fast fuel handling, they have not evaluated the head-end steps considered here There are three general approaches of controlling decay heat: (1) store the fuel for a sufficient decay period so that the fission product heat content is reduced to an acceptable level, (2) dismantle and dissect to an array of the proper size to give a tolerable decay heat, or (3) supply an adequate amount of auxiliary cooling. The first approach (adequate decay time) would incur a large economic plutonium inventory penalty in a commercial fast breeder economy. However, this penalty will not be an overriding consideration in the processing of fuels from the FFTF and early demonstration reactors. Thus, heat emission from these fuels can be resolved very simply by allowing the fuels to decay over a sufficient period (years) of time.

The heat dissipation information given in Sect. 3.2 can be used to evaluate the problems of heat removal from LMFBR fuel, as illustrated in Figs. 4.3-4.7. Figure 4.3 shows the center rod temperature of a hexagonal array of 217 rods as a function of the linear power output of the rods and the emissivity of the rods. Similar calculations for hexagonal arrays of various sizes could be used to determine the size of the array that could be safely handled without exceeding the maximum allowable cladding temperature (Fig. 4.4). The computer code RABBIT can be used to establish the linear power values for all the LMFBR fuels, as given by the example in Fig. 4.5.

The determination of the safe handling time for a particular fuel array can be found from a graph of the form shown in Fig. 4.6. Such a graph is the most conservative estimate of the handling time since the calculations upon which it is based do not take into account any heat

ORNL DWG 69-2392 R I

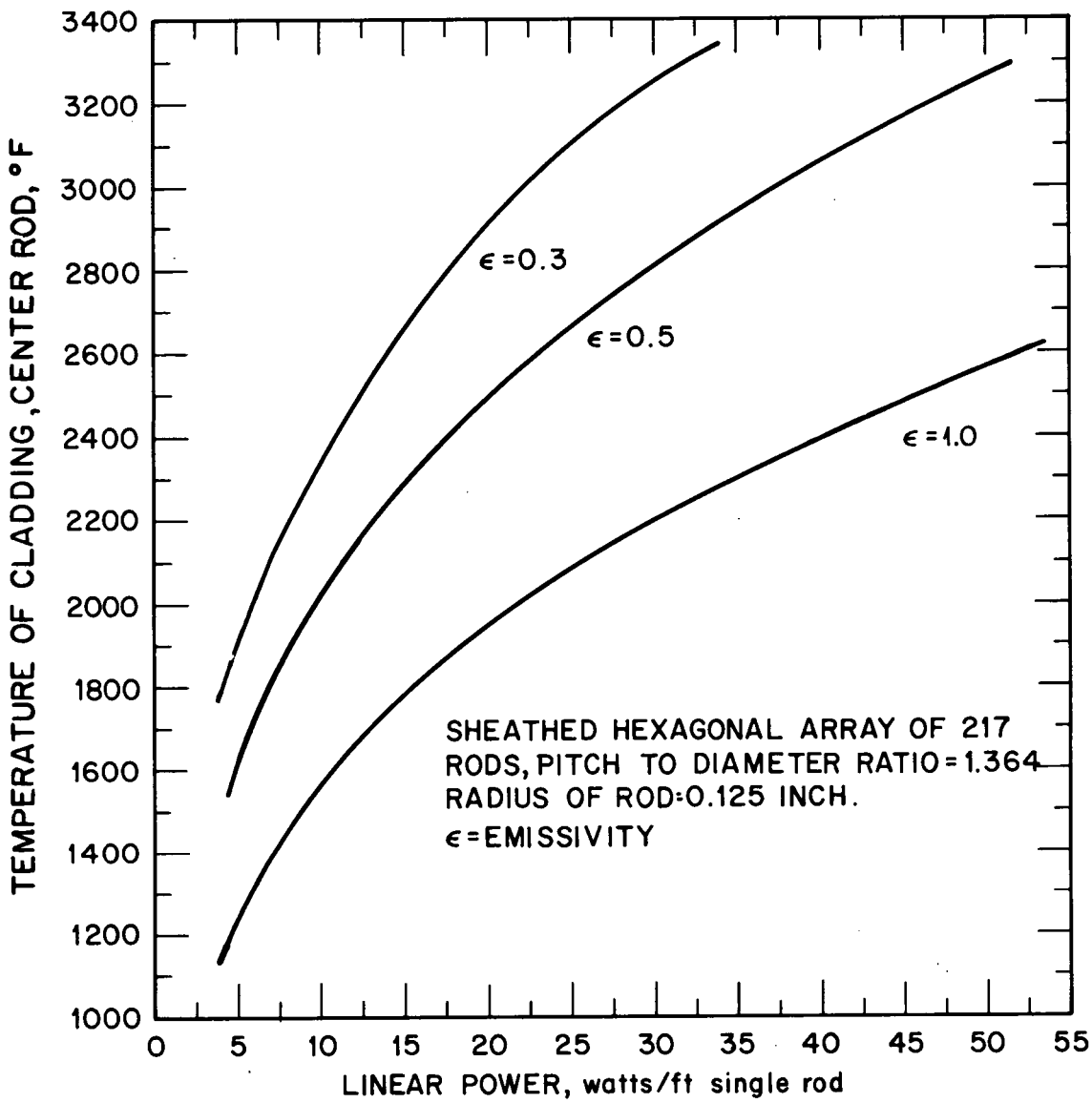

Fig. 4.3. Steady-State Temperature of an LMFBR Core Fuel Rod Array.

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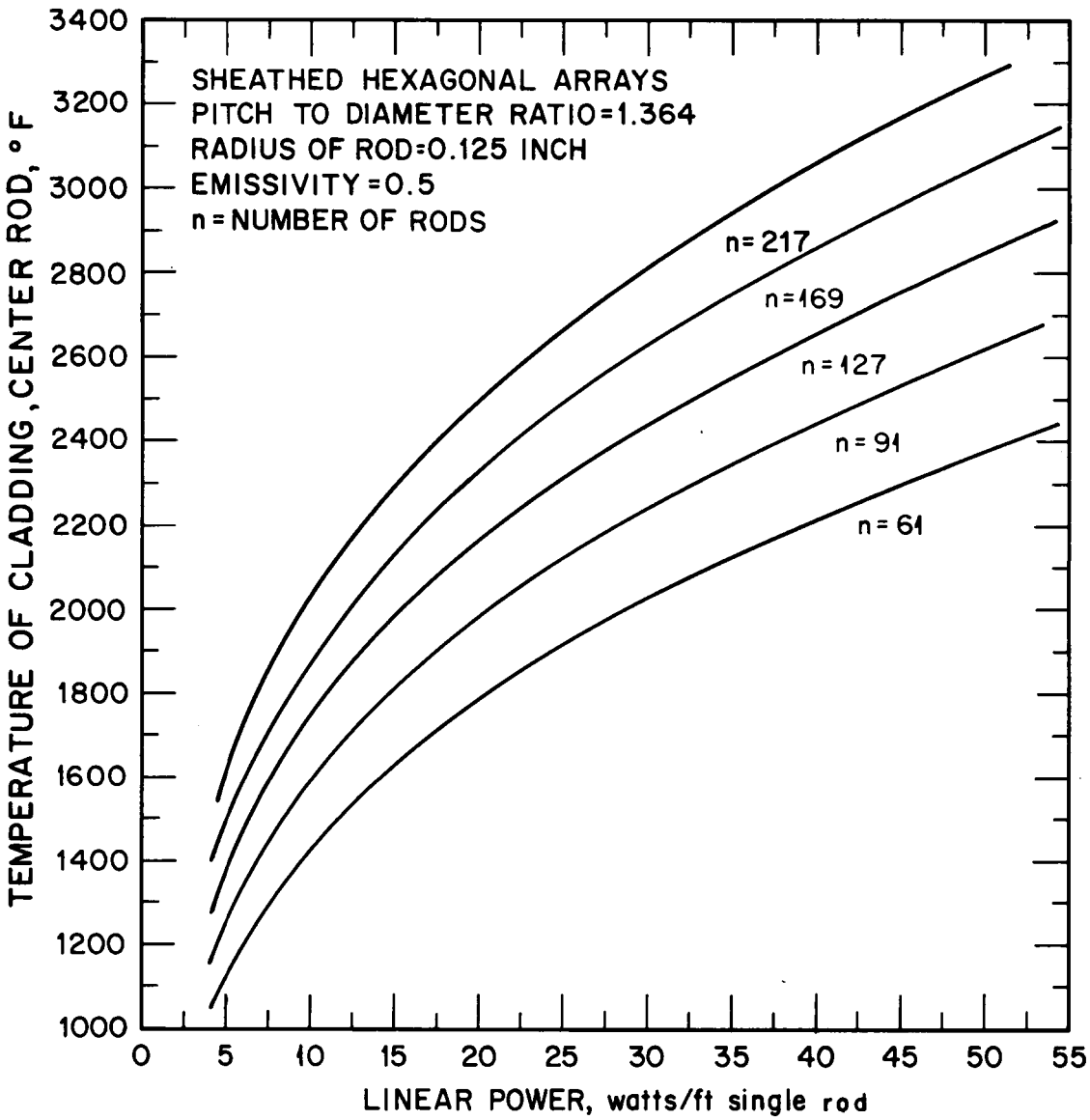

Fig. 4.4. Steady-State Temperature of LMFBR Core Fuel Rod Arrays of Various Sizes.

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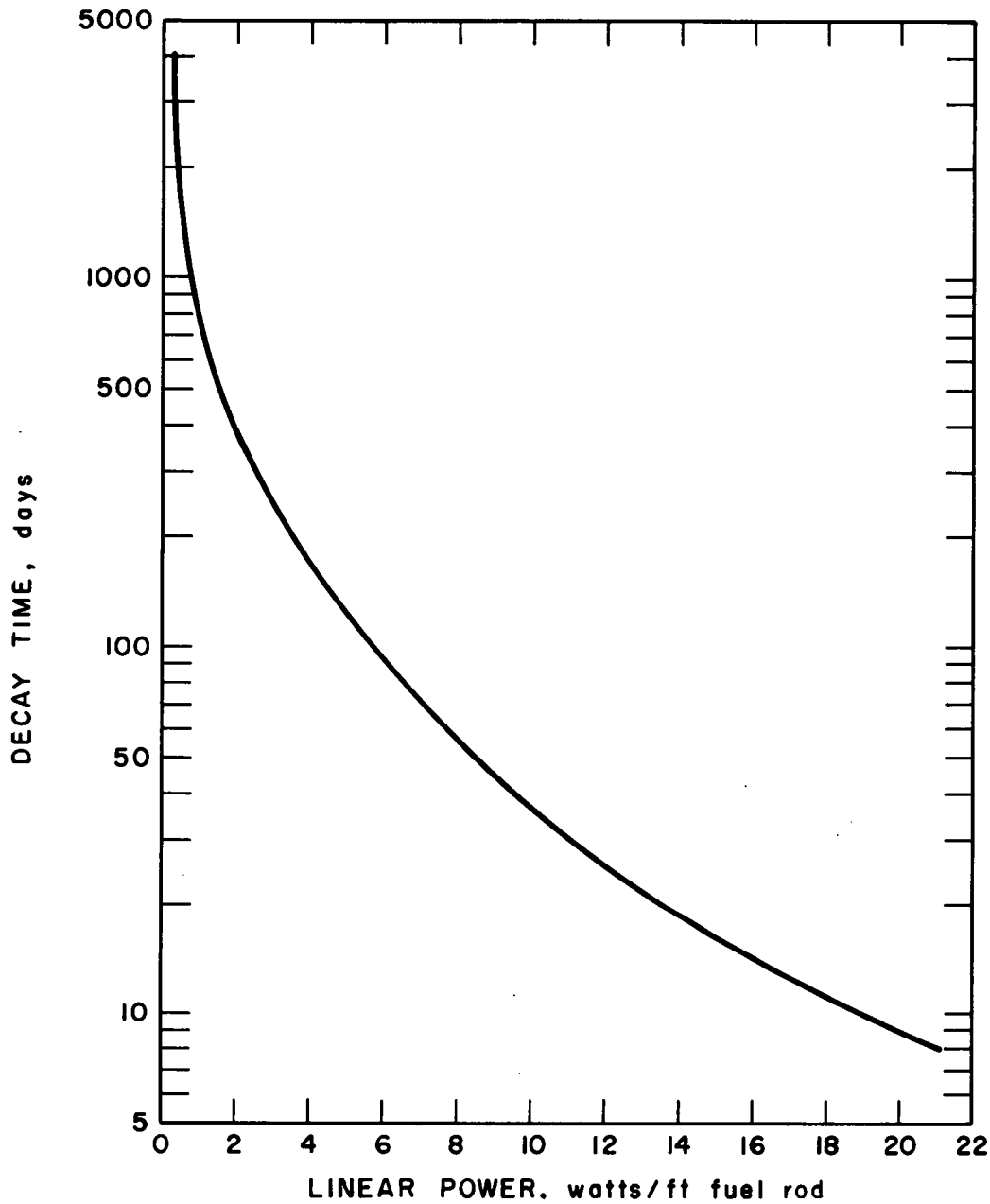

Fig. 4.5. Decay Time vs Linear Power Output of an AI Reference Oxide Core Fuel Rod.

ORNL DWG. 69-2388 R2

CURVE	DECAY TIME, DAYS	WATTS/FT. SINGLE ROD
A	10	18.9
B	21	13.4
C	30	11.1
D	60	7.73
E	150	4.32

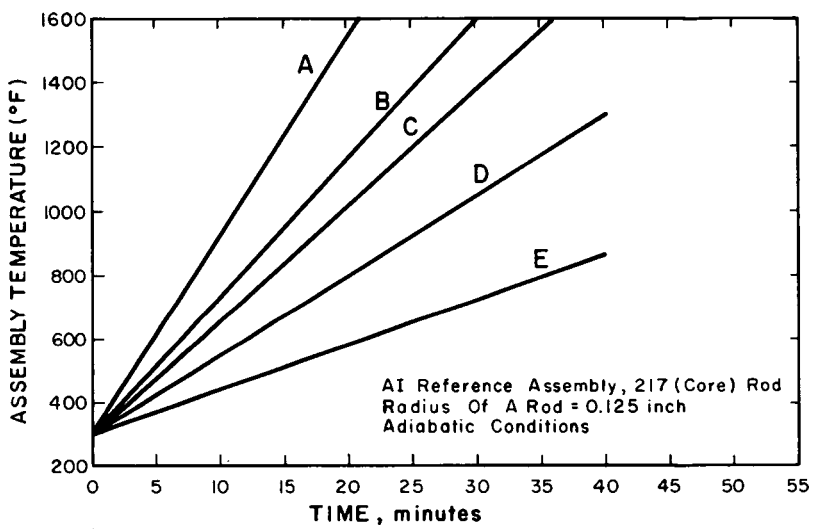

Fig. 4.6. Temperature - Time History of a Typical LMFBR Fuel Assembly.

ORNL DWG 69-2389 R1

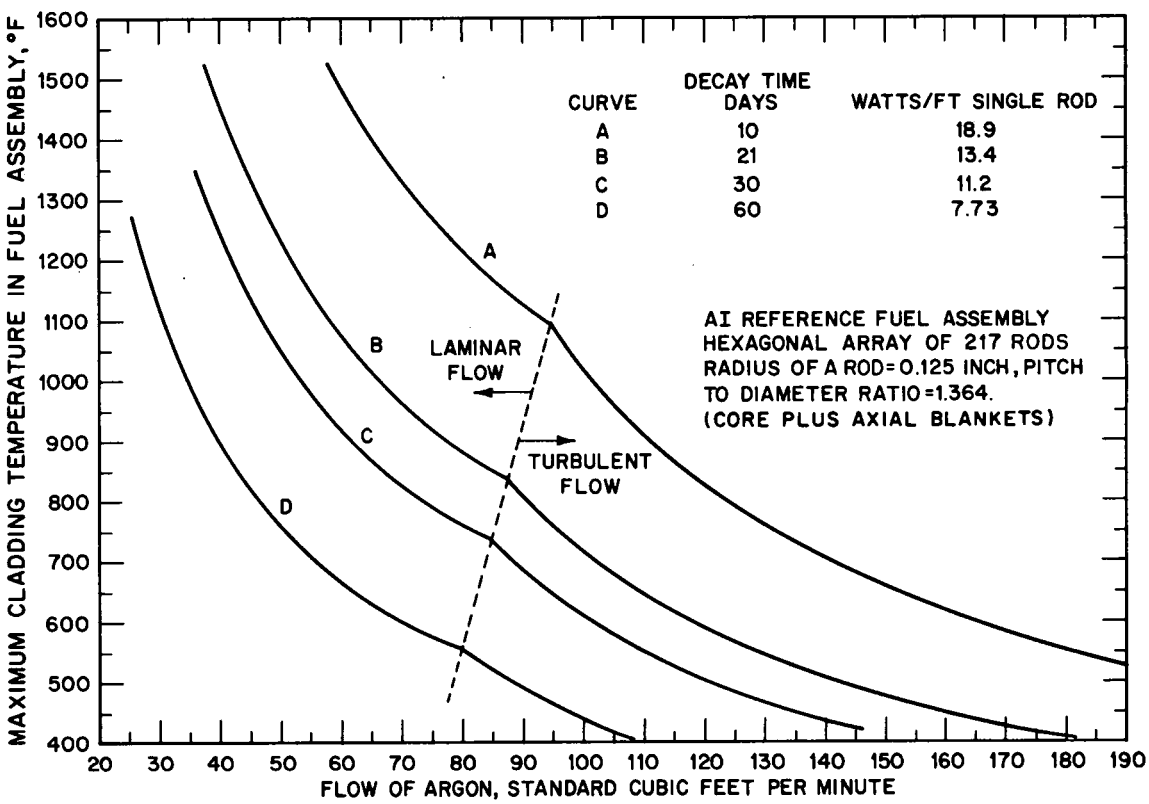

Fig. 4.7. Cooling Gas Requirement for a Typical LMFBR Fuel Assembly.

dissipation from the rods. More realistic time values will be obtained by non-steady-state heat-transfer calculations. Finally, the cooling requirements for an array could be determined from a plot such as that in Fig. 4.7, which gives the maximum cladding temperature of the center fuel rod in the array for various flow rates of cooling gas forced through the assembly. It can be seen from Fig. 4.4 that fuel arrays containing more than 127 rods will require forced cooling, even after 150 days decay (4.32 w per ft of rod), if the temperature of the center rod is not to exceed 1300°F.

Analyses similar to those illustrated here need to be performed for the number of rods contained in various arrays as a function of irradiation and cooling times of interest, and with varying emissivity, to supply the data needed for the evaluation of problems during fuel handling. The dissipation of heat will present a problem in all head-end steps, but it will be most severe in the cleaning, shearing, and voloxidizer steps.

4.3.3 Dismantling of Multitubular Assemblies

The dismantling of hexagonal LMFBR fuel assemblies (4 to 6 in. across flats and up to 18 ft long) and the cropping of the end hardware, along with the slitting of the shrouds, are desirable to rid the downstream processing steps of massive waste metal. In addition, it may also be desirable to further disassemble or dissect a dismantled assembly to aid heat dissipation during the shearing, volatile fission product gas release, and dissolution steps.

The dismantling of multitubular fuel assemblies usually involves one or all of the following operations: (1) decanning, (2) removal of end hardware, (3) displacement of fuel rods from a shroud or sheath, (4) removal of fuel rods from a system of grids, and (5) cutting of metallic strips binding subbundles or subassemblies. A study was made of the FFTF, demonstration, and follow-on reactors to determine if they could be dismantled similarly to LWR reactor assemblies. Included were several cropping and slitting methods and a method for dissection or disassembly, as follows:

(1) Abrasive Cutting - Wet or dry cutting is applicable. Wet cutting is about 20% slower.[25] Dry cutting rates are

normally 10 to 12 in.^{2}/min, and the blade wear is 1:2 (1 in.^{2} of blade per 2 in.^{2} of metal).[26] A holding clamp on either side of the cut is required to hold the work rigid to minimize wheel breakage. Wet and dry wheel speeds are about 8,000 and about 15,000 surface feet per minute (sfpm), respectively. At an optimistic cutting rate of 10 in.^{2}/min on irradiated steel, a cutting time of about 1.1 min would be required to crop a typical fuel radially through the sheath and lower end fittings; about 2 min would be required to slit the 144-in.-long shroud. The wet abrasive disk cropping of LWR fuels has been used routinely at NFS. It could be applied to LMFBR fuels, but a unit will have to be designed and evaluated.

(2) Friction Cutting - A circular steel blade operated at about 22,000 sfpm contacts the work surface, generating heat above 600°C;[25] this reduces the tensile strength of the material, allowing it to be removed easily.

With this type of cutting, the LMFBR assembly sheath would deform before enough pressure could be developed to generate the heat required for effective cutting;[27,28] however, this method could probably be developed for removing the massive end pieces.

(3) Electrolytic Grinding (ELG) - A conductive grinding wheel and an electrolyte (salt-water) flowing between the wheel and the work surface complete an electric circuit which removes the stock metal in the form of a metallic hydroxide.[29] About 90 to 95% of the material is removed by electrochemical action and 5 to 10% by abrasive action; therefore, wheel wear is slight. Approximately 0.1 in.^{3} of metal is removed per 1000 amp per minute. A current density of about 3000 amp/in.^{2} can be attained. It would require 23.5 min to slit the sheath of a typical LMFBR fuel. These slow cutting rates do not make electrolytic grinding attractive for cropping and slitting operations.

(4) Roll Cutting - A special tube cutting device using pipe-cutter wheels that would perform in a manner similar to that of conventional pipe cutters could be adapted to severing LMFBR sheaths. However, the proposed high burnup rates of the LMFBR fuel may negate the use of roll cutters because of the excessive hardness of the irradiated metal.

An operable roll cutting tool has been successfully used in the Idaho EBR-II facility in cutting hex-sheaths from fuels of low burnups. The tool employs spring-loaded cutters at a force of 400 lb in order to follow the irregular surface. Also, cuts are made at a rigid cross section, which provides support for the sheath during cutting.[30]

Longitudinal cuts on the sheath with nibblers and the roll-cutter combination would be difficult on hardened metal; also, the LMFBR shrouds do not appear to have a rigid cross section and would tend to deform rather than be cut.

(5) Other Dismantling Methods - Additional dismantling methods that have been considered include: (1) lixiviation, (2) slitting, (3) torch cutting, and (4) the drilling out of spot-welds, if applicable. None of these methods appear as attractive as abrasive cutting or some other similar techniques. To summarize, abrasive disk cutting, which is currently being used to crop LWR fuels at NFS, appears to be the most attractive method for dismantling LMFBR fuel at this time. However, other simpler, less messy (i.e., involving smaller volumes of metal chips, abrasive dust, coolants) methods are desired. One serious drawback of applying abrasive disk cutting or any other method to LMFBR shroud removal is that it will not be possible to cut completely through the shroud owing to the danger of fracturing fuel rods. In this respect, the deshrouding of LWR fuel is much easier and less complex. The deshrouding and the dismantling of LMFBR fuels are problems of significant magnitude that require extensive design and equipment investigation. There is no doubt that the head-end handling of fast breeder fuel could be

simplified considerably if the fuel rods could be pushed or withdrawn from the sheath.

Some preliminary work has been accomplished using a plasma-arc to deshroud short prototype sections of LMFBR fuel. The process is quite promising but needs additional investigation and evaluation.

4.3.4 Shearing

The shearing requirements for LMFBR fuels appear to be basically similar to those for LWR fuel. However, LMFBR fuel may present two additional requirements: the shearing of intact or broken fuel in the presence of sodium, and simultaneous shearing of broken or intact fuel along with the can surrounding it. Presently, there is no existing technology for the shearing of canned fuel or assemblies or subassemblies in the presence of sodium or other coolants. This is an area that will require substantial development and evaluation. Shearing has been successfully demonstrated at NFS, using Yankee Atomic and Dresden LWR fuels. However, the shearing of high-burnup, short-cooled LMFBR fuels presents problems not normally encountered with LWR fuel. Some of these problems are (1) high rates of decay heat generation, (2) fission gas release, (3) production and containment of very fine core and clad particulates, and (4) the presence of liquid metal coolants. At this time, it appears desirable to have the capability of shearing canned LMFBR assemblies, uncanned assemblies, subassemblies or groupings of fuel rods, and single fuel rods. Applicable techniques and equipment components will have to be developed and evaluated. In order to do this, an extensive research and development program will be required to determine the best blade shape for the moving and fixed blades, type and number of gags, gag forces required to successfully compress the shroud, and methods for restraining and feeding the terminal end of the fuel assembly.

An analysis of mechanical shearing techniques and their relationship to LMFBR fuel, as presently developed, follows.

Assemblies and Subassemblies. - The shearing of unirradiated fuel bundles fabricated with brazed ferrules (3-5/8 in. x 3-5/8 in. - Nuclear

Ship Savannah), crossed wire grids (4-1/4 in. x 4-1/4 in. - Dresden Type IV), or leaf-spring grids ($\leq$ 10 in. x 10 in. - Westinghouse type) as a means of fuel rod retention and spacing has been satisfactorily demonstrated at ORNL. When assemblies fabricated with brazed ferrules or crossed wire grids are sheared, discrete and separate pieces of the fuel rods are produced. In the case of fuel fabricated with grids, chunks consisting of sheared tubing and grid are produced at the grid sections; these chunks are difficult to transfer and handle.

The advantages of shearing an assembly or subassembly are:

1. A large quantity of sheared fuel can be generated using relatively slow-moving equipment.
2. Minimal costly disassembly is required.

There are also several disadvantages:

1. A large shear costs about $250,000.
2. Clamps or gags, which complicate shear design, operation, and remote maintenance, are required.
3. Intermittent shearing is usually required, resulting in a longer residence time for the fuel in the shear magazine. This further complicates heat dissipation problems.
4. Control of sheared batch size is made more difficult.
5. Volatile fission products are released rapidly and in relatively large volumes.
6. Maintenance problems are more complex as the size of the shear increases.
7. The length of the terminal sheared piece usually ranges between 1-1/2 and 3 in. The length of these terminal pieces set the total time required for dissolution.

Multitubular Array. - The General Electric Company proposes to withdraw fuel rods from an assembly and use the approach of shearing an array of fuel rods rather than an assembly in their plant at Morris, Illinois. A satisfactorily demonstrated multirod shear does not now exist; thus, development and evaluation programs will be required to produce a unit that will be useful for LMFBR fuel. This method is attractive for LMFBR fuel,

primarily because of the heat emission problem. However, the technique will probably not be useful for fuels assembled with grids. At ORNL, multirod or single-rod shearing has been demonstrated in a few experiments. As many as 90 porcelain-filled, stainless-steel-clad loose Consolidated Edison Core B type rods (approximately 0.300 in. OD) have been successfully sheared using the 250-ton ORNL prototype shear, and 25 porcelain-filled loose General Electric reference-type rods (0.25 in OD) have been successfully sheared using the small prototype fast-breeder shear without the use of a clamp (or gagging). At this time, it appears that there are no major problems connected with shearing an array of rods; however, there is a maximum number of loose rods (not yet established) that can be satisfactorily sheared without clamping the pieces in position during the time that the cut is being made.

Some advantages of multirod shearing are:

1. The sheared product is free of grids and other hardware. However grids may be present in subassemblies and thus negate this advantage.
2. Good control of batch size is possible.
3. Heat dissipation problems can be minimized by selecting an array of the appropriate size.
4. The amount of fission gas and sodium released per cut can be controlled and minimized by limiting the number of fuel rods sheared.
5. Shear size and cost are reduced to a minimum.
6. The small size of the shear facilitates remote maintenance procedures and allows easy interchangeability.

Some disadvantages are:

1. Individual fuel rods with fragile spaghetti-like properties must be handled, assembled in an array, and placed in a shear magazine.
2. Each fuel assembly must be dismantled and further dissected into individual fuel rods.

Single-Rod Shear. - If LMFBR fuel could be processed as single fuel rods, the problems of heat dissipation and the collection and destruction of liquid metal coolant would be alleviated somewhat. The development of equipment to shear single rods has not progressed to the point that a unit is now being used on spent fuel in the United States. In 1957, work was begun on a single-rod shear at Knolls Atomic Power Laboratory,[31] but it was never completed. The French use a single-rod shear in a small plant at L'Hague, and the Germans are developing one for their pilot plant near Karlsruhe.

A single-rod shear could be used for LMFBR fuel. The advantages and disadvantages are similar to those for the multirod array shear. A conceptional scoping type design that employs rotary blades is presently being prepared. Two rotary shear blades operating at 682 rpm could process up to 5 metric tons of fuel per day; this amount of fuel would be equivalent to 13,736 core and axial fuel rods of the proposed Atomics International Reference Oxide fuel. Such a fuel throughput would require the feeding of about five fuel tubes to each shear per minute and would entail an almost foolproof rapid feed mechanism. Considerable research and development work will be required in order to produce a single-rod shear with a demonstrated practical feed and shear mechanism.

4.3.5 Isolation of the Shearing Operation

When spent nuclear fuel is sheared, iodine, noble gases, and tritium escape from the severed gas plenums and from the sheared faces of the core and blanket. In addition, very fine, easily suspended particles of core and blanket material are produced. The control and collection of these overhead products from the shearing operation in minimal volumes of purge gas are important from the standpoint of cost. If large volumes of gas (air, argon, N_2, CO_2) are permitted to leak into the shear, the resultant off-gas treatment system will be very large and extremely expensive. Further, it will be necessary to isolate the shear from the noncompatible atmosphere existing in subsequent head-end operations such as voloxidation, dissolution, etc. Thus, an important phase of the shear development program involves the development of seals, valves, closures, and operating techniques that will permit close control over the inleakage of gas and the

particulate materials evolved. At this time, data applicable to this situation are meager, if available at all.

Mechanical seals and liquid seals need to be developed and evaluated for this purpose. A successful seal must provide an effective barrier between the cell atmosphere and the shearing atmosphere (inert gas, sodium, etc.) during sémicontinuous or continuous operation. Appropriate leaktight feed magazines for the various forms of fuel to be sheared will also have to be developed as part of the closure problem.

4.3.6 Sodium Handling and Removal

One of the objectives of head-end processing is to prepare the fuel assembly for the nitric acid dissolution step. Since previous work at ORNL and other sites has shown that even small amounts of metallic sodium react explosively with hot nitric acid,[32,33] safety requirements dictate that all of the sodium be either removed or deactivated prior to the dissolution step. Sodium can be introduced into the head-end step of the LMFBR aqueous reprocessing flowsheet in four ways:

(1) as the heat-transfer medium inside the shipping cans, if it is required,

(2) as residual sodium that is present on the external surfaces of fuel assemblies (and shipping cans) after they are removed from a sodium environment,

(3) as the heat-transfer bonding medium inside individual fuel rods, and

(4) as sodium-logged fuel rods (broken, ruptured, or vented).

Gross amounts of sodium used as the heat-transfer medium in shipping cans would probably be recycled back to the canning step instead of being deactivated and sent to waste. If the sodium is not recycled, a method of deactivating gross amounts of this material will have to be established and evaluated. Very little information has been published about the amounts of sodium entrained with the fuel assembly when it is removed from a sodium pool. British workers have reported on the wetting of stainless steels by sodium as a function of oxygen content and temperature of the sodium.[34-36] Estimated quantities of sodium entrained on

EBR-II fuel range from 30 to 60 g (9 to 18 mg/in.^{2});[37] however, most of it is evaporated or blown off by a stream of hot argon prior to the steam deactivation step.[38] Similar data for Fermi reactor fuel range up to approximately 125 g per assembly (25 mg/in.^{2}). Some data on the amount of sodium adhering to a wetted stainless steel surface (~ 10 mg/in.^{2}) were obtained as part of the SEFOR reactor sodium vapor control tests.[39] Data obtained by Van Dievoet et al.[40] show the weight of entrained sodium on a single rod as a function of sodium temperature (13 mg/in.^{2} at 200°C). These tests provide some indication of the amount of sodium we can expect to be entrained on the external surfaces of fuel assemblies. Little information is available concerning the size and nature of LMFBR fuel rod failures and the extent of the sodium logging that may occur in failed rods. Accurate information concerning the amount of sodium associated with an LMFBR fuel assembly regardless of its location (internal or external) must be obtained before meaningful sodium deactivation procedures can be developed. Unfortunately, there is apparently no single operation (including voloxidation) in the head-end steps where all of the sodium can be deactivated and removed. Consequently, both the deactivation methods and the amount of sodium present have to be evaluated for each operation. This would include a determination of the amount of sodium vapor that is tolerable in hot-cell operations, especially with respect to equipment operation. Such information is presently nonexistent and will have to be developed and evaluated. However, we do not propose to duplicate the work of others in the sodium handling field (e.g., that planned at PNL or at the HFEF in relationship to the cleaning, handling, and examination of FFTF fuel). In addition, the Liquid Metals Engineering Center will be a very valuable source of constantly updated information on sodium. Emphasis will be placed only on the problem areas that are unique and restrictive to the head-end processing steps and handling operations. The PNL program and other similar operations will be kept under surveillance so that, as useful information is developed, it can be appropriately applied to our problems.

A survey of the chemistry of sodium[41-48] and of past and present cleaning practices suggests several mechanisms for the deactivation and removal of sodium. These mechanisms are discussed below.

Physical Cleaning Methods: Mechanical Methods: - The configuration of the fuel assemblies precludes the physical removal of sodium by mechanical wiping, brushing, etc. The amount of sodium attached to the surfaces of the assemblies can be reduced by either dislodging it with a high-velocity inert gas stream or flushing it off with a high-velocity inert liquid stream. The efficiency of each method would probably be increased by using an ultrasonic technique.[49] Sodium inside a damaged fuel rod would not be affected. These methods are simple, do not consume reagents, do not require elaborate, expensive cleaning vessels, and, in addition, provide a means of removing fission product decay heat. Whether the amount of sodium present can be reduced to a level such that the location or the intensity of any reaction which might occur during subsequent operations does not present a hazard is open to question. Further studies are needed to ascertain the usefulness of mechanical cleaning methods in a high-capacity processing plant.

Physical Cleaning Methods: Distillation Methods. - Removal of sodium by distillation in an inert gas or vacuum is an attractive cleaning method when the vapor pressure of sodium is greater than 0.01 atm. However, long cycle times, high temperatures (bp of sodium: 882.9°C at 1 atm; 680°C at 0.1054 atm; 533°C at 0.0104 atm), and large, complicated pieces of equipment are required. Vacuum distillation is probably the only method which will remove sodium from inside a ruptured, broken, or vented fuel rod.

Chemical Deactivation Methods. - There are many potential agents for deactivating sodium, but only a few meet the following criteria for a practical reagent:

(1) The reagent should be inexpensive, easily handled, and noncorrosive.
(2) Its reaction rate with sodium must be easily controlled.
(3) It should not produce equally reactive (with water) reaction products.

These criteria eliminate compounds such as the halogens, hydrogen halides, metallic and organic halides, oxygen, hydrogen, ammonia, and many more.

Sodium may be removed by amalgamation with a low-melting metal or alloy. The formation of an amalgam offers the advantages of effective removal of fission product decay heat and a compatible transition to an aqueous processing step. Mercury and lead are two candidate metals. Regardless of the metal used, it will eventually have to be easily and inexpensively separated from the sodium and recycled in the process. In the past, such processes (lead) have been used by the British, but never at production rates sufficiently high for commercial processing plants. Currently, however, the British simply allow the sodium to oxidize during storage over a period of several years.

The general reaction between alcohols and sodium is:

$$ROH + Na \rightarrow RONa + 1/2\ H_2\uparrow \qquad \Delta H_{298} = -48.1 \text{ kcal/mole}$$

The reactivity of sodium with alcohols decreases in order from primary to secondary to tertiary, and as the number of carbon atoms in the alcohol increases. The rate of reaction can be controlled either by regulating the concentration of alcohol in an inert solvent or by choosing a less reactive alcohol. The sodium alcoholate produced could be hydrolyzed with water and stripped from the solvent, which could then be dried and recycled. Some of the problems involved in using alcohol are:

(1) A solvent which does not degrade rapidly when exposed to both high radiation and heat fluxes will have to be found.

(2) The effect of the solvent on downstream processing will have to be minimal or it too will have to be cleaned from the fuel assembly in a second step.

(3) A fire hazard exists with this method due to the presence of large amounts of alcohols plus the hydrogen produced during the reaction.

(4) The waste problem will be complicated by the presence of organics (alcohol plus solvent) in the aqueous NaOH waste stream.

Although the basic water-liquid sodium reaction (see below) is violent, it can be controlled by metering steam into the inert gas atmosphere inside the cleaning vessel.

$$Na(\ell) + H_2O(g) \rightarrow NaOH + 1/2\ H_2\uparrow \qquad \Delta H_{298} = -45.7\ \text{kcal/mole}$$

A water wash after the reaction is complete will remove residual NaOH. Based on work at ORNL, water or steam in carefully controlled quantities will probably not react explosively with carbide or nitride fuels.[51,52] Some problems involved in the steam cleaning method are:

(1) Large amounts of both fission-product-decay heat and reaction heat will have to be removed from the cleaning vessel.

(2) Large amounts of hydrogen will have to be collected and disposed of safely.

Steam cleaning, which is the method used for cleaning EBR-II and Fermi fuels, has been extensively investigated by Atomics International.[53] However, studies of this method will need to be extended and accelerated. This is one of the problem areas which will be investigated at PNL; the data obtained should be of importance to our program.

Problems in Candidate Mechanical Head-End Steps and Alternative Processes due to the Presence of Sodium. - The problems encountered in the principal candidate head-end processing steps due to the presence of sodium are presented in Fig. 4.8. A brief discussion follows:

Cropping (Removal of End-Hardware from a Fuel Assembly). - Uncanned fuel stored in water does not pose a sodium problem in this step. Fuel stored in an inert gas or inert liquid may contain residual sodium on external surfaces and internal sodium in broken or vented rods. Residual sodium on external surfaces can be removed by any one of the several deactivation and disposal methods discussed previously. However, since complete removal of the internal sodium cannot easily be done in a short period of time, the best approach may be to perform the cropping in an inert environment (liquid or gas) and then send the sodium-contaminated, cropped fuel assembly to the next step.

ORNL DWG. 70-2768

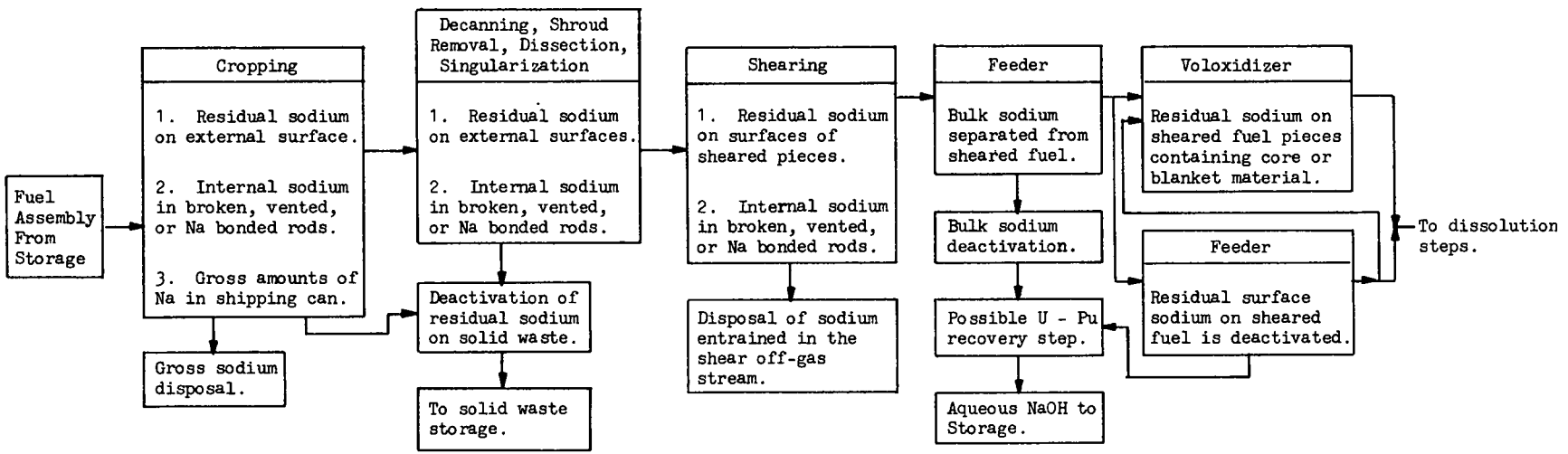

Fig. 4.8. Sodium Problems in the Head-End and Volatile Fission Product Gas Removal Steps.

Canned fuel assemblies have the added problem of bulk sodium (used as a heat-transfer medium) in the cans, necessitating an inert environment for the cropping step. The bulk sodium will have to be decanted, collected, and either (1) deactivated to yield large amounts of aqueous NaOH waste and hydrogen or (2) recycled and reclaimed as metallic sodium. Residual sodium on the surface of the severed end hardware and on the ends of the can will have to be deactivated and removed before these metallic pieces can be disposed of as solid waste.

Decanning, Shroud Removal, Dissection, Singularization, etc. - Sodium can be present in these operations as residual sodium on external surfaces and/or internal sodium in broken, vented, or sodium-bonded fuel rods. Fuel assemblies can be cleaned before or after these operations, although guaranteeing that all of the internal sodium in broken and vented fuel rods has been deactivated is extremely difficult. The other option is to perform these operations in an inert environment, sending the sodium-contaminated pieces to the shearing step. This is particularly attractive when sodium-bonded fuel rods are present. In any case, the residual sodium adhering to the surfaces of the discarded cans, sheaths, etc. will have to be deactivated and removed prior to the solid waste disposal step.

Shearing. - Sodium may enter the shear either as residual sodium on external surfaces or as internal sodium in broken, vented, or sodium-bonded fuel rods. It may leave either (1) as small drops entrained in the shear off-gas stream or (2) with the chopped pieces. While the amount of sodium entrained in the shear off-gas stream may be small, the large volumes of volatile fission product gases also present in this stream will make sodium deactivation and removal exceedingly difficult. The remaining sodium will exit with the chopped pieces to a diverter (not yet designed), which separates the empty gas plenum hulls from the hulls containing core and blanket material.

Cleaning of Empty Hulls. - The empty gas plenum hulls from the diverter may drop onto a vibrating feeder, where bulk sodium will be drained off, collected, and sent to a deactivation and disposal system. (This sodium will be too radioactive to recycle.) The drained hulls will be washed with a water spray before they are sent to the dissolution step. It will probably be necessary to provide a plutonium-uranium recovery step for the aqueous NaOH waste streams.

Volatile Fission Product Gas Removal (Voloxidizer). - The sheared pieces containing core-plus-blanket material will be fed directly to the voloxidizer, where the sodium can be oxidized or heated under controlled conditions. Most of the sodium oxide formed will probably sublime (Na_2O sublimes at 1275°C) and leave with the voloxidizer off-gas. This will make the off-gas collection and disposal problems much more difficult to solve. The Na_2O which remains behind can, hopefully, be passed to the dissolution step without further processing.

Sodium Problems in Alternative Candidate Head-End Processing Methods:

A. Aqueous Decladding - Sulfex

As stated previously, this method cannot be used to process LMFBR fuel; therefore, it will not be discussed.

B. Molten Metal Decladding

If the solubility of sodium in the molten cladding is exceeded, a two-phase mixture containing unalloyed sodium will be formed. This could create problems in the recovery and recycle of the melt metal. For example, the solubility of sodium in zinc is about 3.5% at 575°C. In this case, most of the sodium will probably pass out with the sweep gas.

C. Melt Decladding

The temperatures required, 2750°F or greater, are sufficiently high that the sodium (bp, 1670°F at 1 atm) will volatilize and

exit with a sweep gas, from which it can be condensed and collected.

D. Simultaneous Dissolution of Cladding and Core

Experience at ORNL and other sites has shown that it is mandatory that all sodium be excluded from aqueous dissolution processes if catastrophic explosions are to be avoided. To guarantee the complete removal of sodium from the various LMFBR fuel arrays prior to dissolution is much too difficult to be considered practical.

E. Simultaneous Cutting of Cladding Electrically and Dissolution of Core

The statements made for Method D also apply here.

4.3.7 Conversion of Cladding and Inert Metal Components to a Form Suitable for Handling

As spent LMFBR fuel is processed through the head-end steps, a large amount of inert metallic waste may be generated (e.g., from the cropping of end hardware and the deshrouding operations). In addition, the leaching operation produces a large number of hulls (empty cladding) and spacer wires. The weight ratio of inert metal to the fissile and fuel content of LMFBR-type fuel is about 2 to 1. The ratio for LWR fuel is much more favorable, or about 1 to 1.

The void fraction of the inert metal waste produced from LMFBR fuel is about 80%. It would be highly desirable to drastically reduce this void fraction for conversion in handling, shipping, and storage. However, demonstrated methods and techniques for doing this do not now exist and will have to be developed and evaluated.

A brief analysis of two concepts for converting metallic wastes to simpler forms and reduced volume follows:

1. Melting in an Induction Furnace. - The furnace would be initially provided with a heel of molten metal before it is charged with the various waste metal components. This is necessary to initiate the melting of hollow metallic shapes. The molten waste

product is cast into ingots, thus reducing the metal waste to a minimal volume.

2. Compaction of Metal Waste. - The metallic waste could be compressed by a hydraulic press and the compressed product stored as briquettes. Some preliminary compaction data show that it may be possible to compact the waste to 70% of the theoretical density of stainless steel.

Advantages of the induction furnace concept are: (1) a minimum volume of radioactive waste is produced, and (2) a minimum amount of remotely operated equipment is required.

Disadvantages of the induction furnace concept are: (1) considerable quantities of heat and radioactive fumes are generated, thus requiring an air-cooled furnace room with a suitable off-gas system; (2) the liquid level in the furnace would require a monitoring system; (3) some splattering of molten material would probably occur during the pouring operation, and some slag and scale would be produced when the cooled molds are dumped; and (4) all induction furnaces are provided with ceramic liners which will require replacement from time to time. Presently, a liner can only be replaced manually. Techniques for remote placement will have to be developed and evaluated.

Advantages of the compacting concept are: (1) the mechanical operation is much simpler and cleaner than the induction method, and (2) no radioactive off-gases are produced.

Disadvantages of the compacting concept are: (1) the reduction of the volume of waste material is limited, (2) large quantities of material with various shapes must be handled remotely, and (3) end adapters may be difficult to compact.

4.4 Volatile Fission Product Removal - Task 4

4.4.1 Introduction

This section describes a new head-end processing step, voloxidation, for removing volatile fission product gases prior to aqueous processing

(see Task 4 in Fig. 4.1). The objective of this step is to volatilize and remove the fission product gases iodine, xenon, tritium, and krypton from the fuel. The removal of iodine is of primary importance because of the high retention factors that will be required for ^{131}I for short-cooled fuels. The fission product gases will be disposed of by methods other than by dilution and release to the environment (see Sect. 4.10). Based on past experience and preliminary experiments, it appears that these objectives probably can be met by heating the chopped fuel to some reasonable temperature, less than 750°C, in oxygen or an inert gas stream.

The advantage of using a predissolution step to remove the volatile fission products is that these fission products can probably be released in a carefully controlled off-gas stream. This gas stream will probably consist of a small dry oxygen or inert gas flow from which the iodine, and the rare gases and tritium when necessary, can be simply and effectively trapped.

This task will require the development of an entirely new technology. At this time the principal uncertainties are: obtaining basic chemical data for all forms (including carbides and nitrides) of LMFBR fuel of various Pu/U atom ratios and off-gas compositions; the suitability of various equipment designs such as the fluid bed, the rotary calciner, etc. for carrying out the process; the effect of sodium on the process; and the degree of sensitization of the cladding.

4.4.2 Background Information

A review of information concerning the volatilization of fission product gases shows that it deals principally with UO_2 rather than UO_2-PuO_2. This information is useful, however, when combined with the initial experimental work with UO_2-PuO_2 and the carbides.

The release of fission product gases from irradiated UO_2-PuO_2 fuels is controlled by a diffusion process. The rate of release appears to be enhanced either by oxidation of the fuel or by heating to temperatures up to the melting point.[1] Owing to the high melting points of uranium monocarbide (2375°C), uranium nitride (2630°C), and uranium dioxide (2750°C),[2] the possibility of melting irradiated fuels is not attractive.

Work by others[3,4] and preliminary hot-cell tests at ORNL indicate that fission product gases can be volatilized by heating UO_2 fuel in oxygen. The pieces of sheared fuel (~ 1/2 in. long) were treated with oxygen at 450 to 700°C to oxidize the UO_2 to U_3O_8. The phase change releases interstitially trapped gas and, in general, causes the fuel to break up into fairly small particles. Cyclic oxidation and reduction steps (with hydrogen) can probably be used to prepare uniformly sized material consisting of small particles. Initial results at ORNL indicate that it is probably feasible to release most of the iodine, tritium, and noble gases from UO_2 by this method. Under certain conditions, other fission products (e.g., cesium, ruthenium, tellurium, etc.) have also been released.

In fuel containing 20 wt % or more plutonia, some experimental evidence suggests that the oxidation reaction of urania is suppressed.[4] With the various LMFBR fuels, the feasibility of obtaining a phase change during the voloxidizing step is yet to be established. In limited tests performed at ORNL with fuels containing 20 to 28% PuO_2, no evidence of fuel powdering has been obtained. However, tritium releases from test specimens of sheared, irradiated LMFBR fuel have been encouragingly high (> 90%) in the voloxidizing step. Krypton removal has typically been in the range of 10 to 40% (~ 2000 Mwd/ton), although about 98% was released in one sample of high-burnup fuel (100,000 Mwd/ton). Up to the present time, the lack of short-cooled LMFBR fuel samples has prevented studies with ^{131}I.

The voloxidizer operation will depend on the effects of the properties of the manufactured fuel (chemical form, density, shape, size, temperature and time of sintering, and degree of solid solution); the irradiation history (time, maximum temperature, and burnup); and the chemical reactions and physical processes (temperature, particle size, density after irradiation, oxygen concentration, and time). It will also be necessary to know the effect of mechanical attrition on this process. These variables must be studied as fuel becomes available. In addition, special irradiations of LMFBR fuels are needed to obtain short-cooled materials for the study of ^{131}I behavior.

4.4.3 Effect of Coolants or Bonding Agents Such as Sodium

There is no background information on the effect of liquid metal coolants such as sodium on the volatilization and release of fission gases. Also, the behavior of sodium alone is not known. The acquisition of such information is necessary in the experimental program.

4.4.4 Reaction of LMFBR (Urania, Nitride, and Carbide) Fuels with Oxygen

For urania fuels containing up to about 10% plutonia, the UO_2 is oxidized to U_3O_8 as follows:

$$UO_2(s) + 1/3\ O_2(g) \rightarrow 1/3\ U_3O_8(s) \quad . \qquad (1)$$

The heat of reaction for Eq. (1) is about -25 kcal/mole and is essentially independent of temperature. As stated earlier, there is evidence that the reaction is suppressed in fuel containing 20 wt % plutonia or higher.

The oxidation of UO_2 pellets has been studied in the temperature range of 350 to 575°C.[5] It is reported that, at up to 450°C, the total reaction rate is slow and that, at temperatures above 500°C, sintering occurs even at relatively high gas velocities. Consequently, practical operating temperatures may be limited to the narrow range of 450 to 500°C. Unirradiated arc-melted UC oxidizes slowly below about 600°C, but reacts rapidly between 800 and 1000°C.[6-9] Uranium monocarbide can be made reactive by extended exposure to moist air; the reactive UC ignites and burns in oxygen at about 300°C.[10] Irradiated 20% PuC--80% UC has been burned in 2 hr at 800°C in a stream of oxygen.[11] Powdered, unirradiated UN burns rapidly in oxygen at temperatures above about 450°C;[12] no reference to the oxidation of irradiated UN has been found. Additional oxidation studies will be required, especially with irradiated fuel specimens.

4.4.5 Sensitization of Inert Metal Components of Fuel, and Other Processing Effects

As presently conceived, the release of volatile fission product gases from LMFBR fuel would probably take place in a voloxidizer unit at temperatures from 450 to 750°C in an oxidizing or inert atmosphere. If oxidation is required, the stainless steel cladding, wire spacers, etc. may be burned to the corresponding oxides or become sensitized so that they are extensively corroded or dissolved in the dissolution step. Either situation would produce undesirable quantities of wastes, and the high concentration of iron in the process solution would complicate the solvent extraction process. It will be necessary to study this problem experimentally.

4.5 Dissolution - Task 5

4.5.1 Chemistry

Extensive laboratory and production experience has been accumulated in dissolving uranium metal with HNO_3. Much less information is available concerning the dissolution of UO_2, and less still on the dissolution of UO_2--PuO_2. The chemical equations for the dissolution of UO_2 follow:

$$UO_2 + 2.7\ HNO_3 \rightleftarrows UO_2(NO_3)_2 + 0.7\ NO + 1.3\ H_2O,$$

$$UO_2 + 4\ HNO_3 \rightarrow UO_2(NO_3)_2 + 2\ NO_2 + 2\ H_2O\ .$$

Ordinarily both reactions occur to some extent, with the first tending to predominate at low acid concentrations and the second at acid concentrations greater that 8 M.[1,2] The composition of the dissolver off-gas, therefore, depends on the exact dissolution conditions.

By supplying oxygen and scrubbing with water or dilute nitric acid, most of the oxides of nitrogen can be removed from the off-gas to form HNO_3 for recycle:

$$2\ NO + O_2 \rightarrow 2\ NO_2,$$

$$3\ NO_2 + H_2O \rightarrow NO + 2\ HNO_3\ .$$

Ideally, the dissolution is possible without loss of nitrogen oxides from the dissolver-scrubbing system ("fumeless dissolving"), according to the overall reaction:

$$UO_2 + 2\ HNO_3 + 0.5\ O_2 \rightarrow UO_2(NO_3)_2 + H_2O\ .$$

In laboratory tests,[3] the release of nitrogen oxides from the dissolver solution during dissolution of UO_2 was prevented by adding hydrogen peroxide to the dissolver. In addition, the rate of dissolution of UO_2 in the peroxide--nitric acid solution was greater than in the same nitric acid solution without peroxide.

Both small-scale tests and limited industrial experience have shown that UO_2 can be dissolved completely in 4 to 8 M HNO_3. The rate of the reaction increases with increase in acid concentration and increase in temperature up to the boiling point. The reaction is autocatalyzed by nitrous acid. (Small concentrations of nitrous acid are formed in the solution by the reaction $2\ NO + HNO_3 + H_2O \rightleftarrows 3\ HNO_2$.) Rapid stirring, boiling, or passing a purge gas through the dissolver tends to reduce the nitrite concentration of the solution, thereby decreasing the dissolution rate by as much as a factor of 5.[1]

4.5.2 Dissolution of PuO_2--UO_2

The dissolution of PuO_2--UO_2 in nitric acid is affected by (1) the percentage of PuO_2, (2) the method of fabrication (i.e., mechanically blended oxides, co-precipitated oxides, or sol-gel oxides), and (3) the irradiation level.[4] Pure PuO_2 does not dissolve at a significant rate in nitric acid, and fluoride is required as a catalyst. However, the use of fluoride is undesirable because of corrosion problems. Solid solutions of PuO_2--UO_2 dissolve easily in nitric acid. Thus, maximum dissolution rates and complete dissolution are obtained if the fuel is fabricated initially as a solid solution.[4] ORNL studies[4] demonstrated that mechanically blended PuO_2--UO_2 could be converted to a leachable product by sintering at 1600°C for 16 to 48 hr. Recent German studies[5] with unirradiated fuel demonstrated that mechanically blended 15% PuO_2--85% UO_2 could

be converted to solid solution (< 99.9% dissolved in 8 hr in boiling 14 M HNO_3) by heating the oxide mixture 32 hr at 1500°C, 16 hr at 1600°C, or 8 hr at 1700°C. Studies at ORNL also show that the high irradiation levels at the high temperatures present in fuel rods tend to produce a homogeneous and soluble product. However, fuel in the outer areas of the fuel rods does not always reach sufficiently high temperatures to ensure good solubility.

About 50 small-scale dissolution tests have been made at ORNL with a variety of PuO_2--UO_2 fuels of different preparation histories, plutonium contents, and burnups.[3,6-9] Although inconclusive in many respects, these data show some general trends:

1. The dissolution rate in boiling nitric acid increased considerably with increase in acid concentration in the range of 4 to 13 M.
2. The initial dissolution rate of essentially all samples in boiling 8 M HNO_3 was high; more than 90% of the plutonium was dissolved within 2 to 3 hr. A small fraction (usually 1 to 10%) of the plutonium was more refractory and dissolved very slowly.
3. Sol-gel and co-precipitated oxide samples generally gave less than 0.2% plutonium loss to the residue after leaching for 8 hr in boiling 8 M HNO_3. Results were more erratic and recoveries were usually poorer (up to 8% loss) with mechanically blended oxides.
4. The amount of stainless steel cladding dissolved by leaching for several hours in boiling 8 M HNO_3 was usually in the range of 1 to 10% of the total cladding. The dissolution rate was initially rapid but then decreased, indicating that the cladding surface had been sensitized, probably primarily as a result of the loss of chromium from the surface to the sodium coolant.

The available leaching data indicate that it will be advantageous to dissolve the major fraction (> 90%) of the fuel in a short leach treatment with, for example, 8 M HNO_3 and to provide for some retention in the dissolver (or recycle to the dissolver) of the residue in order to afford a longer dissolution time for the more refractory fuel fraction. With this

provision, it appears that an average fuel residence time of 4 hr should be sufficient to obtain essentially complete plutonium recovery from most fuels. However, many more leaching experiments with a variety of LMFBR fuel samples of different compositions and histories are needed to confirm this and to better define the range of dissolution rates and metal recoveries that can be expected. If extended leaching with nitric acid still leaves excessive amounts of plutonium in the residues from some fuel samples, these residues, which should represent only a small percentage of the total fuel weight, could be given an additional leach with HNO_3--HF to dissolve all of the plutonium.

Dissolution of PuO_2--U_3O_8. Assuming that the oxidative heat treatment is used to remove gaseous fission products from the PuO_2--UO_2 fuel prior to dissolution (see Sect. 4.4), the fuel fed to the dissolver will be PuO_2--U_3O_8. Dissolution of oxidized irradiated fuels has been studied briefly at ORNL. It appears that the leachabilities of these fuels are not appreciably affected by the oxidation treatment, but more data are needed for a variety of conditions to confirm this. The amount of fumes produced should be only one-third the amount resulting from dissolution of UO_2:

$$U_3O_8 + 6.7\ HNO_3 \rightarrow 3\ UO_2(NO_3)_2 + 0.7\ NO + 3.3\ H_2O\ ,$$

$$U_3O_8 + 8\ HNO_3 \rightarrow 3\ UO_2(NO_3)_2 + 2\ NO_2 + 4\ H_2O\ .$$

No fumes result from the reaction with PuO_2.

$$PuO_2 + 4\ HNO_3 \rightarrow Pu(NO_3)_4 + 2\ H_2O\ .$$

4.5.3 Iodine Behavior in Dissolution

Elemental iodine is evolved readily from hot nitric acid solutions. Any iodine initially present in the solution as iodide ion oxidizes rapidly to elemental iodine. Any initially present as iodate is reduced to elemental iodine by the nitrous acid formed during the dissolution of UO_2.

The tendency, therefore, is for essentially all of the iodine to be driven off from the dissolver during the dissolution operation, particularly if there is a large air flow through the dissolver. In process practice, the dissolver usually is coupled with either an updraft or a downdraft reflux condenser. This hinders the removal of iodine since much of the iodine condenses in the condenser and eventually is washed back into the dissolver by the refluxing solution.[3,6,8] Prolonged refluxing of the iodine in this manner, after the dissolution reaction has stopped and nitrous acid is no longer being formed, can lead to conversion of a significant fraction of the iodine to the iodate form.

It is evident from the above discussion that the amount of iodine separated from the fuel during dissolution can vary widely, depending on the exact arrangement of the dissolver-condenser system and the manner in which it is operated. Production plants have reported that 30 to 95% of the iodine is evolved during dissolution. In small-scale dissolution tests at ORNL, removal of more than 99% of the iodine has been accomplished by appropriate arrangement of the dissolver-condenser system and by adding iodide and nitrite to the system at the end of the dissolving period.[8] The equipment arrangements and the procedures used, however, may not be attractive for large-scale use. Rather than make a special effort to remove iodine during dissolution, it appears preferable to utilize present production equipment arrangements (i.e., dissolver coupled with a downdraft condenser) and techniques and to remove iodine from the solution during feed adjustment (Sect. 4.6.2). Data from operating plants indicate that iodine in the gas stream leaving the condenser will be efficiently removed when this stream is scrubbed with dilute acid to remove nitrogen oxides. Depending on the amount of iodine present, it may be desirable to sparge the scrub solution to remove iodine before the solution is recycled to the dissolver.

Addition of Mercury. - The mercury-iodine complexes are so stable that the addition of mercuric ion to the dissolver greatly decreases the amount of iodine evolved during dissolution even at boiling temperatures.[10] This treatment has been used in production plants[10,11] to help control iodine

release during the processing of short-cooled fuels. However, mercury volatilizes and can be troublesome during fixation of high-level wastes. In addition, the mercury-iodine complex is extracted by TBP[12] and is later found to some extent in the product strip solution. For these reasons, addition of mercury to the dissolver solution does not presently appear attractive. Essentially quantitative removal of iodine from the aqueous feed appears to be preferable in order to prevent the introduction of iodine into the solvent extraction system.

4.5.4 Dissolution of Carbide and Nitride Fuels

In laboratory studies at ORNL, unirradiated UC was found to dissolve in boiling 2 to 16 M HNO_3 within 10 hr, yielding an off-gas composed mainly of NO_2, NO, and CO_2 and a dark, red-brown solution containing uranyl nitrate, soluble organic species, and probably also some HNO_2.[13-16] Between 30 and 50% of the carbide carbon was converted to the organic species; the remainder was converted to CO_2, which was evolved. Although all the organic species have not been identified, oxalic and mellitic (benzene hexacarboxylic) acids are products of the reaction.[14-16] Dissolution of UC in boiling 13 to 16 M HNO_3 required about 6 hr, with the consumption of about 8 moles of acid per mole of carbide.

Uranium monocarbide specimens (4.4 to 5.0% C) that had been irradiated to burnups of up to 6000 Mwd/ton dissolved in about 4 hr in boiling 15.8 M HNO_3, yielding solids-free dark solutions 0.5 to 1 M in $UO_2(NO_3)_2$ and 7 to 11 M in HNO_3.[17] Similarly, PuC (1000 Mwd/ton) and 20% PuC--80% UC (20,000 Mwd/ton) dissolved in boiling 13 M HNO_3 in 4 hr, yielding solids-free green-black solutions.[17]

The soluble organic species must be largely destroyed to prevent severe emulsification and complexing of heavy-metal ions during solvent extraction.[17,18] For UC fuels a reasonably satisfactory method for doing this consists of refluxing the product solution [e.g., 0.5 M $UO_2(NO_3)_2$--11 M HNO_3] for an additional 6 hr to partially oxidize the organic species. The excess acid is then removed by distillation (little additional oxidation of the organics), the distillation residue is diluted to Purex feed-solution composition [1.36 M $UO_2(NO_3)_2$--3 M HNO_3], and the solution is

made 0.2 M in $KMnO_4$ and refluxed for 2 hr. The MnO_2 formed is removed by filtration. With both unirradiated and irradiated UC, about 5% of the original carbide carbon remained in the feed solution after the $KMnO_4$ oxidation step; however, most of the species deleterious to solvent extraction were eliminated. In experiments with irradiated PuC and UC--PuC, the solutions after $KMnO_4$ treatment contained 25 to 32% of the carbide carbon in the form of organic compounds. These compounds had no adverse effect on extraction of the feed solutions; however, after stripping, 0.4% of the plutonium was retained in the solvent. Thus, some of the residual organic impurities may have been extracted into the solvent, where they complexed and retained plutonium during stripping.

Pyrohydrolysis, followed by dissolution of the resulting oxide in HNO_3, appears to be a more promising process for dissolving carbide fuels than direct dissolution in HNO_3. However, the work on PuC and UC--PuC was limited to a very few hot-cell experiments, and it is possible that some better means of oxidizing the organic species can be found. Reaction of the carbides with boiling water, followed by dissolution of the oxides in nitric acid, was an attractive process when tested with unirradiated materials, but this scheme is not feasible for irradiated carbides since they are inert to boiling water.[19-21]

Nitride Fuels. - Unirradiated UN dissolved readily in 0.5 to 15.8 M HNO_3, yielding an off-gas composed of N_2, NO, N_2O, and sometimes NO_2 (8 to 16 M HNO_3), and a yellow solution containing uranyl, ammonium, and nitrite ions.[22,23] The exact quantities of the various nitrogen products are a function of the acid concentration, reaction temperature, and time of the reaction. In addition, the nitrite ion is in equilibrium with gaseous NO, so these quantities can vary with the experimental conditions. Ammonium ion appears to be an intermediate in the oxidation of the nitride nitrogen to free nitrogen. It is more stable at the lower acid concentrations and lower temperatures, but continues to oxidize slowly after all of the uranium has dissolved. This is not a serious problem since the ammonium ion is readily destroyed by increasing the acid concentration and/or temperature. For example, less than 0.3% of the nitride nitrogen was found as ammonium ion after heating an 8 M HNO_3 solution for 1 hr at 80°C.

Optimum dissolution conditions for UN will depend upon the physical state of the fuel. It would be hazardous to contact sizable quantities of powdered UN with 8 to 16 M HNO_3 because of the very rapid gas evolution. Use of acid concentration in this range might be safe, however, when dissolving pellets. Preliminary scouting tests showed that pellets of irradiated UN dissolved in boiling 8 to 16 M HNO_3 (but not in 4 M HNO_3), leaving an unidentified black residue.[23] No solvent extraction data on UN product solutions are available.

No information could be found in the literature concerning the dissolution of unirradiated or irradiated PuN or UN--PuN specimens in nitric acid.

4.5.5 Equipment and Operating Procedures

The nature of LMFBR fuels occasions some special requirements for the leaching system. The high-burnup, short-cooled fuel will introduce heat dissipation problems[24] that are much more severe than previously encountered, especially if storage is required prior to leaching. Criticality and off-gas considerations will have a greater controlling effect than heretofore on the design and operation of the leaching system. Also, if a volatile fission product removal step (Sect. 4.4) is used, the fuel fed to the dissolver may be in a finely divided condition and wholly or partly separated from the cladding. The possibility of shearing complete fuel assemblies because of internal grid arrangements may occasion other requirements for handling random-shaped chunks of fuel.

Generally, dissolution equipment under consideration for LMFBR fuel falls into two categories--batch and continuous. As far as is known at this time, various designs of either type of leacher could conceivably be scaled in size to accommodate the plant capacities currently in mind (e.g., 5 metric tons U + Pu per day as the oxide). Normally, batch equipment would be duplicated to produce a smoother plant operation since dissolution could take place in one unit while the other was being dumped and reloaded. Aside from this refinement, a single line of equipment appears to be possible for the capacity

mentioned. Additional studies to more accurately determine dissolution kinetics, especially of highly irradiated fuel, are required for final design considerations.

Batch leaching is the only method used at the present time on a commercial basis. Its technology has been extensively developed for both unirradiated and irradiated fuel. Nuclear Fuel Services, Inc., West Valley, New York, and Eurochemic, Mol, Belgium, have successfully demonstrated its reliability. In batch leaching, sheared fuel (contained in a basket) and dissolvent are contacted batchwise in a leacher. After the fuel is dissolved, the product solution is withdrawn for further processing. The leached hulls, after washing and inspection, are packaged for disposal. Usually, the leacher basket is attached to the shear for filling, and transferred to the leacher either directly or via a storage area. Alternatively, the sheared fuel can be transported to the leacher via a chute. The use of a basket for containing sheared fuel affords a convenient means of transporting the leached hulls after dissolution.

The advantages of batch leaching are:

1. It is a method that has been demonstrated to be reliable; the absence of moving parts minimizes maintenance.
2. The baskets provide a means of surge-storage (this may not be an advantage for LMFBR fuel).
3. Sealing to minimize air inleakage and to control off-gases is relatively simple.

Disadvantages of batch leaching include:

1. Large surges of fission product gases and nitrogen oxides are released to the off-gas system.
2. Usually, provision must be made to handle a variety of basket types.
3. Criticality considerations may dictate the use of rather small baskets and, therefore, the handling of many baskets.

4. This scheme is not compatible with the goal of a low-inventory, high-throughput processing unit having a short turnaround time.

Experience with batch leaching on a large scale has been limited to uranium metal, UO_2, and UO_2--ThO_2.

Continuous leaching has been partially developed at ORNL, using an enclosed rotary spiral leacher for unirradiated UO_2.[25,26] The main obstacles to successful development of a continuous leacher are:

1. A method of introducing solids to, and removing them from, a leacher in a continuous manner while maintaining a vapor seal is required. This may require use of liquid mechanical seals.

2. Efficient transport of all solids without regard for shape and size must be accomplished.

3. The reliability of the equipment must be satisfactorily demonstrated.

Advantages of continuous leaching include:

1. Evolution of fission product gases and nitrogen oxides to the off-gas system is more uniform.

2. Basket handling is eliminated, as is the storage of sheared fuel and the attendant heat dissipation problems.

3. The possibility exists for operating a single line of equipment with no downtime for loading and unloading.

4. It is compatible with the goal of having a high-throughput, low-inventory processing unit.

5. Product concentration variation would occur gradually, not stepwise, allowing better control of subsequent processes, especially if they are closely coupled.

Criticality considerations for LMFBR fuel leaching equipment are strongly dependent on whether the plutonium remains distributed uniformly throughout the fuel mass and/or solution or whether it segregates

as a precipitate or powder in one location of the dissolver. It has been estimated that, for no plutonium segregation, pipe-type dissolvers up to 15 in. in diameter could be used with 0.2 M Cd as a soluble poison.[27] For complete segregation, diameters of 5.5 in. would be the limiting size. Leacher concepts for LMFBR fuel formulated to date have generally favored the conservative approach, since no guarantee exists that the plutonium will not segregate. Using the conservative approach, the allowable size of the unit is not affected by the plutonium concentration in the fuel being leached. The use of 3-in. annuli without soluble poison, and 4-in. annuli with soluble poison, around a central poisoned core of at least 8 in. seems to be a reasonable approach at this time.[28] As criticality calculations become more sophisticated, a more liberal treatment of concepts can be made. This would take the form of reducing central poisoned core diameters and allowing increased fuel throughput.

Fuel dissolution to a solution strength of 1 M heavy metals or higher is desirable from the standpoint of high leacher throughput and of producing a solution that does not require evaporation prior to solvent extraction. The problem of designing a critically safe evaporator for LMFBR feed solutions is a formidable one. Feed solutions 1 M in heavy metals will require only simple dilution for acid adjustment prior to use of the dilute (~ 0.25 M metal in feed) Purex flowsheet. A dissolver product rate of 3.6 gpm at a heavy metals concentration of 1 M is equivalent to 5 metric tons of fuel per day.

A new batch leacher concept exists that calls for two basket-type leachers, each operating with an 8-hr filling period (during which dissolution occurs), a 4-hr heel dissolution period, and a 4-hr hull rinse and cleanout period. The use of two such leachers allows a steady fuel feed to be maintained and minimizes storage lag time. In addition, simultaneous basket loading and dissolution minimize off-gas surges. This concept should more accurately be classified as semicontinuous. Uncertainties in this concept that require additional developmental efforts include the design of a semicontinuous filling system with

vapor seal, a more efficient basket withdrawal and hull dumping method, a means of withdrawing and transporting insoluble wastes from the dissolver bottom, and the establishment of a forced liquid flow pattern to ensure dispersion and fluidization of particulate matter. In accordance with the conservative criticality criteria, each blanket leacher for a 5-metric-ton-per-day facility would have a 4-in. annulus about a 25-in.-diam poisoned core, would be 8 feet high, and would dissolve fuel to a 1 M total heavy metal content in the presence of 0.2 M Cd soluble poison or its equivalent. Elimination of the soluble poison (thus requiring 3-in. annuli), operation to produce a dilute product solution, or the realization of less favorable dissolution kinetics would decrease the leacher capacity. The result would probably be the use of multiple units since the leacher described above is already quite large.

Concepts for the continuous dissolution of LMFBR fuels are many and varied. Several uncertainties are common to all and would require additional developmental efforts. For units handling sheared fuel, these include the design of a feeding and vapor seal mechanism, the design of a hull discharge seal, and a means for discharging finely divided insoluble wastes. For units handling powdered fuel that has been separated from the hulls (e.g., after voloxidizer treatment), the problems are the designs of (1) a powder feeding and vapor seal mechanism, (2) a means for discharging insolubles, and (3) an effective foam control system.

Perhaps the most promising continuous dissolver concept is the rotary screw. A single 25-in.-diam unit with a 4-in. annulus and poisoned core should be suitable for a 5-metric-ton-per-day facility, assuming an average fuel dissolution time of 4 hr. This choice of leaching time assumes that provision would be made for recycling the dissolution heel (or otherwise providing extended leaching time for the small fraction of the fuel that is relatively refractory). Having three sections, each with 14 screw flights on a 6-in. pitch, this unit would operate with intermittent rotation and thus allow a semicontinuous feeding arrangement. A fourth unit in series would serve to rinse hulls. Restriction to a 3-in. annulus may mean having additional units

in series in order to obtain the necessary residence time. These screw-type leachers would have to be constructed using a perforated sheath around the screw to contain hulls and spacer wires, yet allow liquid circulation to the heated stationary housing. Also, provision to agitate particulate matter that passes through perforations, either by liquid flow or mechanical brushes, must be made. A prototype unit must be operated to establish confidence in performance.

If the volatile fission product removal operation yields fuel that is separated from the hulls to the extent that a mechanical separation of fuel powder (PuO_2--U_3O_8) and hulls is feasible, dissolution could be accomplished by directly dissolving the fuel powder and washing the hulls. Since fuel powder can be brought into immediate and intimate contact with the dissolvent, the residence time required should be much less, possibly only one-fourth that for the direct leaching of sheared fuel. Hull washing could occur in a rotary-screw dissolver as described above. A likely method of dissolving the fuel powder would be to use a thermal recirculating pipe-type dissolver similar to that developed at Idaho.[29] A unit fabricated from 6-in. pipe having a 30-ft recirculating length would dissolve 5 metric tons of fuel per day. As before, unfavorably slow dissolution kinetics or more strict criticality regulations might necessitate the use of multiple units. The success of this type of dissolver is dependent on its capability for maintaining the liquid flow velocity in the upflow region at a value that will retard the downward movement of the powder sufficiently to give the desired powder residence time, yet not high enough to result in carryover of an undue amount to the product take-off point. The quantity of gases generated during dissolution becomes quite critical to the operation of this type of equipment and may restrict its usefulness. Since this equipment is characterized by a rather short residence time, accurate knowledge of the dissolution kinetics of the irradiated fuel is required. Success may be strongly influenced by the degree of powder separation. Fuel chunks or agglomerates would probably be undesirable.

A second type of powder dissolver that has merit is a rotary-screw unit. Assuming favorable dissolution kinetics, a unit with a capacity of 5 metric tons of fuel per day would have a 3-in. annulus around an 8-in.-diam poisoned core, and would be 10 ft long. This type of dissolver would effectively handle fuel chunks and agglomerates and would be relatively insensitive to the quantity of off-gases produced.

Dissolution of Carbide and Nitride Fuels. - The only available data on the dissolution of carbide and nitride fuels are those obtained from laboratory studies. Most of these studies were made with uranium--thorium rather than uranium--plutonium fuels. Leaching of these fuels is expected to differ from oxide fuel leaching in residence times required and methods of off-gas control rather than in differences in design of the leaching equipment. The leaching data are too sparse to allow estimates of the capacities of different types of leachers.

4.6 Feed Preparation - Task 6

The aqueous feed discharged from the dissolver will contain solids (undissolved fission products, corrosion products, etc.) and, depending somewhat on the contactor used, may require clarification prior to solvent extraction. Preparation of the feed for solvent extraction also will include adjustment of the plutonium valence and nitric acid concentration, and probably, a treatment to remove iodine.

4.6.1 Adjustment of Acid Concentration and Plutonium Valence

After dissolution of the PuO_2--UO_2 fuels in 4 to 8 M HNO_3 (Sect. 4.5), the desired metal concentration and nitric acid concentration (3 to 4 M) in the feed can be obtained by diluting this concentrated solution with 0.5 to 1 M HNO_3.

Adjustment of the Plutonium Valence. - The dissolution of LMFBR oxide fuels produces a mixture of Pu(IV) and Pu(VI); for best results, the Pu(VI) should be reduced to the more extractable Pu(IV) prior to extraction. The amount of Pu(VI) formed during dissolution increases with an increase in boiling time and ferric iron content of the solution,

but typically ranges from 10 to 50% of the total plutonium.[1-4] Apparently the radiation power density of the solution can affect the valence of the plutonium. Feed solutions prepared from short-cooled, high-burnup LMFBR fuels conceivably could have radiation power densities in the range of 5 to 50 ($\beta + \gamma$) w/liter, depending on the heavy metal concentration and whether the core is processed separately or in combination with the blanket.

Various chemical reagents have been used for reducing Pu(VI) to Pu(IV). Nitrite ion has been used most extensively:[3-6]

$$PuO_2^{2+} + NO_2^- + 2\ H^+ \rightarrow Pu^{4+} + NO_3^- + H_2O\ .$$

Nitrite ion is provided by adding a salt such as $NaNO_2$ or by bubbling NO or NO_2 gas through the solution. The reaction is rapid at 50 to 60°C. As part of the sol-gel program at ORNL, plutonium in nitric acid solutions has been reduced quantitatively to the quadrivalent state with NO gas at about 60°C. Even with concentrated ^{238}Pu solutions, which had radiation power densities of about 50 w/liter, quantitative conversion of the plutonium was obtained by this method.[7]

Hydrogen peroxide also has been used to reduce Pu(VI) to Pu(IV), but results have been erratic, particularly if appreciable concentrations of ferric iron are present.[3,6] British workers[8] have also noted that reduction of Pu(VI) to Pu(IV), at least with hydrogen peroxide, becomes more difficult as the radiation power density of the feed is increased.

Other investigators[2,9,10] have accomplished the conversion of Pu(VI) to Pu(IV) by, first, reducing part of all of the plutonium to Pu(III) with U(IV) or Fe(II) and, then, oxidizing the Pu(III) to Pu(IV) with nitrite ion.

In studies[1] at ORNL, the plutonium in LMFBR fuel (irradiated test specimens) solutions with a beta-gamma activity of about 3 w/liter was adjusted to greater than 97% Pu(IV) by adding $NaNO_2$ to the solution and heating briefly at 50°C. There was no apparent change in the

plutonium valence state when the solution was allowed to stand for one week.

Thus it appears that chemical methods are available for adjusting the plutonium valence. Use of nitrite ion, supplied as NO or NO_2 gas, is possibly the most attractive method since it adds no contaminating metals to the process waste. However, further analysis of the applicability of these methods on a plant scale is required.

4.6.2 Removal of Iodine

Separation of essentially all of the iodine from the fuel prior to solvent extraction is desirable since it would simplify the treatment of off-gases from the solvent extraction and subsequent processing operations. The iodine concentration in the final dissolver solution is expected to be relatively low since most of the iodine should have been removed during the shearing, voloxidation, and dissolution treatments. The iodine remaining in the solution may be present primarily in the form of iodate, which cannot be removed by sparging with air. However, iodide or nitrite compounds can be added to reduce the iodate to elemental iodine, which can then be sparged from the solution. A large fraction of the iodine in the dissolver solution may be evolved during adjustment of the plutonium valence with nitrite, particularly if the nitrite is supplied by bubbling NO gas through the solution. This might be coupled with the addition of some cold iodide ion to the solution to provide additional reducing power as well as to provide carrier iodine for increasing the removal of radioiodine.

4.6.3 Feed Clarification

High-burnup LMFBR fuel will contain large amounts of fission products. The solubilities of some of these (e.g., ^{95}Nb) will be exceeded, particularly if concentrated feed solutions are produced. In dissolution tests at ORNL with LMFBR fuel test specimens of 20,000- to 73,000-Mwd/ton burnup, the residue weights varied widely but usually were in the range of 1 to 10% of the initial fuel

weight.[11,12] Appreciable amounts of ruthenium and rhodium, along with silica and stainless steel corrosion products (especially chromium), were found in the residues. These residues settled fairly rapidly in the fuel solution. However, filtration through Whatman No. 42 filter paper was extremely slow. Production facilities such as Hanford have been able to process unclarified feed solutions in pulsed solvent extraction columns with the organic phase continuous. This may also be possible with LMFBR solutions. Alternatively, the feed could be clarified prior to solvent extraction.

Centrifuges have been used to clarify feed solutions at the Savannah River Plant. Although the use of filters is a possibility, it is not particularly attractive. Cyclone separators were used to clean up the fluid of the aqueous homogeneous reactor at ORNL.[13]

More information is needed concerning the amounts and the physical nature of the residues before a choice of the most suitable clarification method can be made. At this time, the use of centrifuges or hydroclones is favored. A bank of thirty 0.25-in.-diam. by 1.5-in.-long hydroclones in three stages of ten each could conceivably separate micron-sized particles from a dissolver solution flowing at the rate of 3.6 gpm.

4.7 Solvent Extraction - Task 7

For more than 20 years, solvent extraction processes, particularly those using tributyl phosphate (TBP) extractant, have been used preponderantly, not only in the United States but also throughout the world, for recovering uranium and plutonium from nitric acid solutions of irradiated thermal reactor fuels. The first process (Redox[1,2]), employed hexone (methylisobutyl ketone) as the solvent. Later, processes using TBP extractant (Purex process[1-4]) and dibutyl carbitol (Butex process[1,2]) were developed. The Purex process has some important advantages over the Redox and Butex processes and has superseded them. In the United States it is used in government-owned plants at Hanford, Savannah River, and Idaho Falls. It is also used in the first U.S. commercial

power reactor fuel processing plant, which is operated by Nuclear Fuel Services, Inc. (NFS), in West Valley, New York. The General Electric Company is planning installation of a processing plant near Morris, Illinois, to recover low-enriched fuel materials irradiated up to 40,000 Mwd/ton in thermal power reactors. This plant will use the Purex process in the main processing cycle. The Allied Chemical Corporation has also announced plans for a fuel processing plant that will use the Purex process. The process is presently being used in foreign processing plants, including those in England and France, and at the Eurochemic facility at Mol, Belgium.

4.7.1 Purex Process

Extensive laboratory studies of the Purex process and the extensive experience acquired in production plants have resulted in substantial improvement of both the process and equipment over the 20 years that they have been in use. Extensive data are available for the extraction of Pu(IV), Pu(VI), uranium, and fission products over a wide range of aqueous-phase compositions and TBP concentrations.[4-6] In addition, the development and application of TBP extraction processes in the recovery and purification of elements other than uranium and plutonium, for example, thorium (Thorex process[1]), have added substantially to the TBP extraction technology.

A typical arrangement of the first-cycle Purex process flowsheet is shown in Fig. 4.9. Prior to extraction, the plutonium in the aqueous feed is adjusted to the quadrivalent state, its most extractable form. The uranium and plutonium are extracted into solvent consisting of 30 vol % TBP in a hydrocarbon diluent. This step separates them from the bulk of the fission products which remain in the aqueous phase and are discharged to waste. The extract is scrubbed with dilute nitric acid to increase the separation from fission products, and the scrubbed solvent phase then passes to the partitioning section where the plutonium and uranium are separated. This is done by contacting the solvent with a reducing agent such as ferrous sulfamate to reduce the plutonium to nonextractable Pu(III),

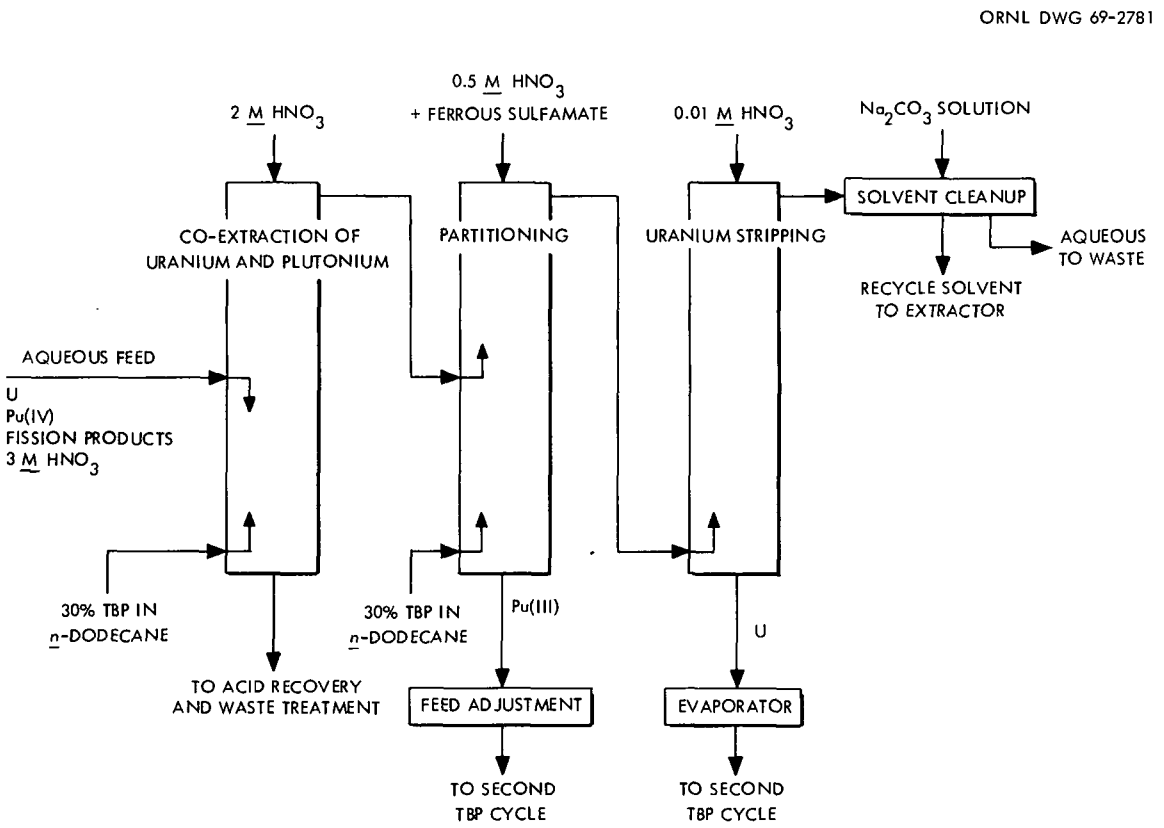

Fig. 4.9. First-Cycle Purex Extraction-Partitioning Flowsheet.

which transfers to the aqueous phase. Some of the uranium also transfers to the aqueous phase and is backscrubbed into the solvent phase with fresh solvent. Finally, the uranium is stripped from the solvent phase with very dilute nitric acid. At this point the plutonium and uranium products have been decontaminated from gross fission products by factors of 10^3 to 10^4. The strip product solutions are then sent to additional purification cycles (see Sect. 4.8).

An alternative to partitioning in the first cycle is to co-strip the uranium and plutonium with dilute nitric acid. These metals can then be co-extracted and partitioned in a second TBP cycle.

Flowsheet changes to Accommodate LMFBR Fuels. - Adaptation of the Purex process to the treatment of LMFBR fuels will require some changes in the usual process flowsheet, owing primarily to the relatively high Pu/U ratio in these fuels:

1. The high plutonium content of the fuel presents criticality problems. Criticality can be controlled by use of a dilute flowsheet (5 g of fissile plutonium in the feed, 15% TBP), as proposed by Nicholson.[7] The 15% TBP flowsheet was recently demonstrated successfully at ORNL in cyclic batch tests with simulated fuels and with feeds prepared by leaching irradiated LMFBR test specimens.[8] The average fuel burnup was about 40,000 Mwd/ton, and the estimated solvent radiation dose per cycle was about 0.3 whr/liter. Alternatively, soluble and/or fixed neutron poisons can be added to the system to allow use of the usual more-concentrated flowsheet. A choice between these methods will probably be made primarily on the basis of engineering considerations (e.g., criticality, equipment capacities, see Sect. 4.7.3) since it should be possible in either system to obtain high recoveries of plutonium and uranium, and efficient separations from fission products, by appropriate choice of solvent/aqueous flow ratios.

2. Because of the relatively large concentrations of plutonium in this system, the recovery of plutonium from LMFBR fuels is more sensitive, as compared with that from thermal reactor fuels, to the concentration of metal in the solvent. At ambient temperatures, the system must be operated with relatively low solvent loadings (80 to 90% of those ordinarily used) in order to avoid excessive plutonium reflux and high plutonium losses. Operation at these lower loadings would decrease the obtainable decontamination factors somewhat. Limitations on solvent loading are less at 40 to 50°C because of the more efficient plutonium extraction at these temperatures.

3. Because of the large amounts of plutonium present, use of ferrous sulfamate in the partitioning cycle contributes relatively large amounts of sulfate and iron to the system. This is undesirable and emphasizes the need for using a different reductant.

4. The large amounts and large total value of plutonium in the system dictate that increased emphasis be placed on achieving essentially complete recovery of plutonium in the solvent extraction circuits. By comparison, the uranium will be of negligible value, at least for many decades. Because of these considerations, the optimum arrangement of the Purex flowsheets for treating LMFBR fuels may be appreciably different from those developed for thermal reactor fuels.

Under comparable contacting conditions, the processing of the highly radioactive LMFBR fuel solutions will result in increased formation of solvent degradation products (due to increased absorption of radiation by the solvent) as compared with the amounts formed in processing low-burnup thermal reactor fuels (which constitute essentially all of the past Purex processing experience). Excessive formation of degradation products can lead to serious process problems. However, recent assessments of this potential problem indicate that, even during treatment of 30-day-cooled, high-burnup materials, radiation damage to the solvent should be low enough that effects on the

process efficiency should not be serious, particularly if fast contactors are used. Extensive experimental work is needed, however, to confirm this tentative conclusion.

In processing LMFBR fuels, the tendency to form plutonium polymer will be increased considerably because of the relatively high plutonium concentrations. Since the polymer is not extracted, its presence in the extraction system could lead to high plutonium losses. In addition, the polymer has a relatively low solubility and can precipitate, leading to criticality hazards. Once formed, the polymer is difficult to decompose. The tendency to form polymer increases with decrease in the acidity of the aqueous phase. Although conditions for polymer formation have been well-defined (at least at 25°C) for laboratory conditions,[9,10] careful consideration must be given to engineering of the process to avoid conditions that could lead to polymer formation. For example, dripping of water condensate into the feed solution could produce a localized area of low acidity and result in formation of polymer.

Partitioning Reductants. - In the Purex process, separation of plutonium from uranium is accomplished after co-extraction by reducing the plutonium to selectively strip it from the solvent (Fig. 4.9). Ferrous sulfamate is used as the reductant in thermal fuel processing plants in the United States. However, LMFBR reactor fuels have a much higher plutonium content than thermal fuels, and consequently the use of ferrous sulfamate would contribute relatively large amounts of iron and sulfate to the waste. The sulfate would also interfere severely with plutonium extraction in the second TBP cycle.

The use of ferrous nitrate, stabilized with a small concentration of a holding reductant such as hydrazine, has shown promise in small-scale cold tests.[11] Its use has the advantage of eliminating the troublesome sulfate, although it still contributes iron to the waste. By appropriate control of the acid concentration, the partitioning can be accomplished by using as little as 25% of the stoichiometric amount of iron needed to reduce all of the plutonium. This

amount of iron does not have a serious disadvantage from the waste standpoint since it is equivalent to only about 20% of the weight of the fission products. The ferrous nitrate solution is easily prepared by reacting ferric nitrate with metallic iron and is adequately stable. Partitioning with ferrous nitrate--hydrazine solutions is presently being evaluated at Hanford on a production basis.

Uranium(IV) has been studied extensively[12] as a reductant by many investigators and is used on a production basis in France. It has the advantage of not adding metal contaminants to the waste. The U(IV) nitrate can be prepared by electrolytic reduction,[13] by reduction with hydrogen in the presence of a platinum catalyst, or by photoactivated reduction with formaldehyde.[14] The U(IV) nitrate solution is usually stabilized with hydrazine. As with ferrous nitrate solutions, effective partitioning with U(IV) can be accomplished by using considerably less than the stoichiometric amount of U(IV) needed to reduce all of the plutonium.[15]

Based on present knowledge, there is little to choose between U(IV) and ferrous nitrate as partitioning reductants. The former has an advantage from the waste standpoint but is somewhat more difficult to prepare and to apply. The reaction rate for reduction with U(IV) is apparently slower than with Fe(II), which may be of importance if short-residence contactors are used.

Reducing the plutonium by contacting the solvent extract with hydrogen gas and dilute nitric acid in a column containing a platinum catalyst has shown promise in preliminary small-scale tests.[16] German workers have reported[17] success in reducing the plutonium electrolytically in a cell containing the mixed phases; porous porcelain barriers are used to separate the electrodes. These techniques need more study to determine if they are competitive with the U(IV) and Fe(II) reduction methods.

Disposition of Uranium. - It seems likely that, for many decades, makeup uranium for the LMFBR fuel cycle will be derived from the stockpile of gaseous diffusion tailings rather than by recycle of uranium

from LMFBR fuel processing plants. The latter uranium will probably simply be stored until needed. The uranyl nitrate product solution from the first cycle may be sufficiently decontaminated that it can be evaporated and stored as concentrated uranyl nitrate solution or as the hydrated solid. Alternatively, the first-cycle product solution can be concentrated and processed through additional TBP cycles to achieve the purity that would allow either easier interim storage or direct return to the reactor fuel cycle.

4.7.2 Solvent Damage in the Purex Process

It is known that TBP decomposes slowly in acid solution (by dealkylation and hydrolysis) during processing to form principally dibutylphosphoric acid (DBP), along with much smaller amounts of monobutylphosphoric acid (MBP), and that the rates of these reactions are enhanced in a high radiation field. The DBP and MBP, when present at sufficient concentrations, decrease decontamination from fission products and increase uranium and plutonium retention by the solvent during stripping. They can contribute to the formation of "cruds" and emulsions which, under extreme conditions, can eventually cause shutdown of the system. The normal Purex flowsheet includes an alkaline wash in each cycle to remove DBP and MBP from the solvent and to prevent their accumulation to the level where such process problems are encountered.

Fuel solutions derived from high-burnup, short-cooled fast reactor fuels may have radiation power densities ranging from 5 to 50 ($\beta + \gamma$) w/liter, depending on the heavy metal concentration in the aqueous feed and on whether the core is processed separately or in combination with the blanket. The question has been raised as to whether problems arising from solvent irradiation damage effects may preclude successful application of the Purex process to treatment of these highly radioactive solutions. There are two additional factors relating to potential solvent damage that must be considered and which have been of negligible importance in past Purex experience with thermal reactor fuels:

1. The relatively high concentration of plutonium, particularly ^{238}Pu, in the organic extract provides a continuing source of radiation after the extractant leaves the extraction section.

2. When a significant concentration of radioiodine is present in a feed obtained by dissolving short-cooled fuel, the iodine may accumulate in the organic phase, giving it a high background activity level and causing further degradation.

The radiation-enhanced degradation of process solvents has been studied at many laboratories. Papers presented at the Solvent Extraction Chemistry Symposium[18] in Gatlinburg, Tennessee, in 1962, and at the International Conference on the Solvent Extraction Chemistry of Metals[19] at Harwell, Great Britain, in 1965 give good accounts of many of these studies and include extensive bibliographies. Many of these data have, in turn, been summarized by Salomon and Lopez-Menchero.[20]

Blake made an extensive survey[21] of the literature relating to solvent radiation damage and came to the following main conclusions:

1. Most of the previous laboratory tests are of little or no use as a basis for predicting radiation damage effects in actual fuel processing. This is because, to obtain measurable effects, a laboratory exposure frequently consisted of one massive dose, which was well in excess of that anticipated during each cycle of LMFBR fuel processing. In addition to the possibility of causing secondary radiation effects, the "one-shot" exposure eliminates possible ameliorating effects that may result each time the solvent passes through the normal extraction-scrub-strip-cleanup cycle while accumulating the same total dose in a continuous process. Further, whereas it has been established that degradation proceeds by the chemical reaction of organic and aqueous phase components and that the extent of reaction is enhanced by radiation, many of the laboratory solvent irradiation studies have been made without the aqueous phase present.

2. In most of the early tests, kerosenes and "naphthas," as typified by "odorless kerosene" and Amsco 125-82, were used. These diluents (containing branched and cyclic paraffins) were considered to be essentially inert during most of these experiments. The fallacy of this opinion has been established, and the more-stable n-paraffin hydrocarbons are now favored for process use. It appears likely that many of the test results with acidified organic phases, particularly those obtained at the high irradiation dose levels, were appreciably influenced by the presence of degradation products of the diluent as well as by degradation products of TBP. Thus, the extrapolations of test results obtained using these diluents to the much more stable TBP--n-dodecane system, which will almost certainly be used in processing LMFBR fuel solutions, are not always meaningful.

Small-Scale Cyclic Tests. - To more closely approximate plant operations, researchers at Windscale[22] set up a continuous recycle experiment in which the solvent phase was contacted with feed solution in a ^{90}Sr radiation field and then subjected to scrubbing, stripping, and cleanup. The solvent continued to perform satisfactorily, even after an accumulated exposure of 20 whr/liter (0.125 whr liter^{-1} cycle^{-1}).

Recently at ORNL, three batch cyclic tests were completed in which the solvent, in contact with $UO_2(NO_3)_2$--HNO_3 solution, was irradiated by a ^{60}Co source in increments of 0.3, 1.0, or 2.0 whr liter^{-1} cycle^{-1} until the integrated radiation doses reached 6 to 10 whr/liter.[23] In all of the tests, including characterization tests on samples of the irradiated solvent, phase separations were rapid and sharp, with no evidence of any solids formation. The 0.3-whr-per-liter-per-cycle exposures had no observable effect on the solvent. Stripping of plutonium and uranium and extractions of ruthenium and zirconium were essentially identical to the results obtained in comparative tests with unirradiated solvent. At exposures of 1.0 or 2.0 whr liter^{-1}cycle^{-1}, the only observed changes were an increase in the extraction of ^{95}Zr

and a slight increase in retention of uranium by the irradiated solvent. Although the data were somewhat scattered, the average zirconium extraction coefficient obtained with unirradiated solvent was 0.013. This compared with average coefficients of 0.011, 0.027, and 0.036 when the exposures per cycle were 0.3, 1.0, and 2.0 whr/liter, respectively. At the latter two exposure levels, the zirconium extraction coefficients did not increase with cycling, indicating that the degradation products (presumably mostly dibutylphosphoric acid) responsible for the increased zirconium extraction were removed effectively by the alkaline wash treatment. Although slight retention of uranium by the solvent after stripping was detectable in the tests in which the per-cycle exposures were 1.0 and 2.0 whr/liter, this was of negligible importance since, at all radiation levels, the amounts retained were less than 0.1% of the total uranium.

Cyclic tests similar to the ^{60}Co tests also were made at ORNL with feed solutions prepared by leaching irradiated LMFBR fuel specimens. Some ^{90}Sr was added to one of the feed solutions to increase its radiation density.[24] No deleterious effects on solvent performance were detected through five complete cycles at a solvent dose level, per cycle, of about 0.3 whr/liter.

Production Experience. - Although the single-cycle doses received by process solvents at the Savannah River and Hanford plants are not accurately known and are probably relatively low, the cumulative doses received are undoubtedly high. For example, the Savannah River plant uses a TBP--*n*-paraffin solvent phase that has been sufficiently recycled to accumulate doses of tens of watt-hours per liter;[25] yet the uranium--fission product decontamination factors remain high, and physical performance of the solvent is satisfactory. This is significant from two standpoints: (1) degradation products of TBP are being removed in the washing cycle to a low concentration level, and (2) degradation products of the diluent, which are not expected to be removed in the solvent washes, have not accumulated to levels that seriously impair decontamination.

The results obtained at ORNL during the processing of irradiated, short-decayed thorium fuel by the Thorex process have often been cited.

The feed solutions in these runs had radiation densities of about 10 whr/liter.[26] However, operational problems, not necessarily associated with solvent radiation damage, were encountered in these tests; consequently, results of this test program cannot be extrapolated to predict the solvent damage effects that would occur in the processing of fast-reactor fuel solutions.

Possibly the most convincing demonstration of satisfactory sustained operation in which the solvent received a relatively high single-cycle dose (~ 1.4 whr/liter) was in processing fast-reactor uranium-molybdenum fuels at Dounreay. In a paper[27] by R. H. Allardice of Dounreay describing this operation, it was stated that:

> "Active operation of the D.E.R.E. Fast Reactor reprocessing plant on irradiated fuel which has produced a feed activity of > 2 curies $\beta\gamma$ activity per milliliter has not resulted in any insuperable effects on the plant solvent which is tributyl phosphate in odorless kerosene. At these levels the solvent, although exposed to a calculated radiation exposure of 1.4 watt-hours per liter per pass does not retain fissile material and does not markedly influence either the decontamination performance or the physical behavior of the first cycle. The solvent does retain fission product γ activity but to an extent which does not cause any operational difficulties, apart from protecting the operators from γ radiation. Consistent solvent wash decontamination factors are obtained and there is no detectable difference in physical behavior and properties between fresh-washed solvent and plant solvent."

Behavior of Iodine. - In the processing of LMFBR fuels, it will be important to remove most of the iodine from the fuel in the process steps prior to solvent extraction in order to facilitate control of radioiodine release from the processing site (see Sect. 4.10). However, since significant amounts of iodine may still be present in the solvent extraction feed, obtaining a better understanding of its potential behavior in the solvent extraction system is important. The extractability of the iodine by Purex solvent depends on the particular species present; elemental iodine is readily extracted, while the iodide and iodate species are not

extracted. No significant amount of iodide ion should be present in the feed solutions since it oxidizes readily in nitric acid to produce elemental iodine.

It is known that the extracted iodine reacts fairly readily with the solvent to form organic iodides that are not removed in the solvent clean-up treatment. In a laboratory test[28] at ORNL, only 20% of the ^{131}I that had aged in a TBP--n-dodecane extract for one week could be stripped with a sodium carbonate--sodium thiosulfate solution, whereas more than 90% was stripped by this solution 1 hr after the extraction. Workers at Dounreay[29] found that about one-half of the iodine was extracted from 88-day-cooled uranium-molybdenum fuel solution. Essentially all of that extracted was retained by the organic phase after scrubbing, stripping, and alkaline washing of the solvent. When the solvent was saturated with stable iodine prior to use, the rate of buildup of radioactive iodine was decreased.

An ORNL estimate indicates that the radiation dose resulting from the accumulation of radioiodine in the solvent would not increase the per-cycle solvent radiation dose obtained in pulsed columns by more than 30% over that obtained if no iodine were in the system. This estimate assumes that no more than 1% of the iodine present in 30-day-cooled LMFBR fuel would be present in the aqueous feed to solvent extraction.

Solvent Dose Estimates. - ORNL estimates (Sect. 4.7.6) indicate that in processing 30-day-cooled mixed core and blanket material in pulsed columns, the solvent radiation dose would be about 0.1 whr liter^{-1} cycle^{-1}. The alpha radiation dose from plutonium accounts for less than 10% of the total dose. The estimates assume use of a relatively dilute flowsheet (7 g Pu/liter in the aqueous feed, 15% TBP solvent) in geometrically unrestricted equipment. Considerably higher doses would result from processing more concentrated feeds, particularly those derived from dissolution of core material alone. In any case, however, it appears that the solvent dose during processing in pulsed columns should not exceed 0.5 whr liter^{-1} cycle^{-1} and would probably be considerably lower than this value. In fast contactors such as the Savannah River centrifugal contactor and the ORNL stacked-clone contactors, the solvent dose would be 5

to 10 times lower than in pulsed columns (Sect. 4.7.6). Assessment of the potential importance of solvent-radiation damage in LMFBR fuel processing appears, therefore, to entail definition of single-cycle effects and cumulative effects with cycling on solvent performance when the dose is in the range of 0.05 to 0.5 whr liter^{-1} cycle^{-1}. On the basis of available information, it appears that these effects should be small. Additional experiments, especially engineering-scale experiments with highly radioactive feeds, are needed to confirm this conclusion and the validity of the solvent-dose estimates.

Solvent Cleanup. - Dibutylphosphoric acid (DBP) and monobutylphosphoric acid (MBP), the principal degradation products of TBP, are removed efficiently from the solvent by scrubbing the latter with a solution of sodium carbonate or sodium hydroxide.[18,19,22] The alkaline treatment decomposes metal salts of these acids [e.g., $UO_2(DBP)_2$, $ZrO(DBP)_2$, etc.] to produce the sodium salts, which distribute almost quantitatively to the aqueous phase. Efficient contact of the phases, usually at elevated temperatures, must be attained for maximum efficiency. The Savannah River plant uses a turbomixer of special design to ensure good contact. The alkaline wash is often coupled with a dilute acid wash and filtration to increase the efficiency of fission product removal. Naylor[30] has summarized the treatment procedures used by a number of domestic and foreign plants (Table 4.3).

Alkaline washes are not effective for removing certain diluent nitration products or TBP nitration products that retain fission products, particularly zirconium-niobium and ruthenium.[18,19] A number of treatment methods, including scrubbing with alkaline solutions containing complexing agents or contacting with solid adsorbents such as activated alumina, zirconia, silica, charcoal, and bentonite, have been used with varying degrees of success to remove the fission products from the solvent. These methods, however, are usually not effective for removing the offending diluent degradation products which again give trouble in the subsequent cycle. In recent years the kerosene and naphtha-type diluents have been replaced in many processing plants by n-paraffin hydrocarbons (primarily n-dodecane) diluents. The latter are much more stable, and the importance of problems

Table 4.3. Solvent Washing Techniques Used in Purex-Type Processes

Plant	Solvent Wash Steps				
	1	2	3	4	5
Eurochemic, Belgium	Filtration	0.25 M Na_2CO_3 in mixer-settlers	Filtration		
Marcoule, France	0.5 M Na_2CO_3 in last mixer of backwash contactor	Filtration through 10-to-20-μ sintered stainless steel	0.5 M Na_2CO_3 in packed column (60°C)	1 M HNO_3 in packed column (25°C)	Filtration through 8-μ sintered stainless steel
La Hague, France	Na_2CO_3 (70°C)	NaOH	Na_2CO_3	Filtration	1 M HNO_3 in mixer-settler
Windscale, U.K.	0.01 M HNO_3 in 4-stage mixer-settler (25°C)	0.1 M Na_2CO_3 in Holley-Mott contactor (60°C)	0.1 M NaOH in 4-stage mixer-settler (60°C)	0.01 M HNO_3 in 2-stage mixer-settler (25°C)	
Hanford, U. S.	0.23 M Na_2CO_3 + 0.05 M $KMnO_4$ in a special contactor (50°C)	0.32 M HNO_3 in a pulsed column (50°C)	0.24 M Na_2CO_3 in a turbomix contactor (40°C)		
Savannah River, U. S.	0.24 M Na_2CO_3 in a turbomix contactor (55°C)	Dilute acid in a turbomix contactor (25°C)	0.38 M NaOH in a turbomix contactor (55°C)	Filtration through a bed of fiberglass	
NFS, U. S.	0.2 M Na_2CO_3 in a column	0.2 M Na_2CO_3 in a 2-stage mixer-settler	0.01 M HNO_3 in a column		

associated with diluent degradation products has greatly diminished.[18,19] The net effect of this has been to simplify the treatment procedures needed for efficient solvent regeneration.

Steam and flash distillation have been suggested as possibilities for providing efficient purification of degraded solvents.[31-33] Distillation is apparently too expensive to be used after each cycle but might be used periodically for purifying the solvent.

In processing fuel solutions containing significant amounts of iodine, appreciable accumulation of iodine in the solvent in the form of organic iodides can be expected, as discussed previously. No attractive method for removing these from the solvent phase is known. The ^{131}I, which is the principal iodine isotope of concern, would eventually, of course, tend to reach a steady-state concentration in the solvent since it is continually lost by decay.

The solvent treatment methods presently used in processing plants provide adequate cleanup of the solvent so that efficient product decontamination is achieved routinely. The number and sequence of treatment steps vary from plant to plant; however, at present, there is no good basis for choosing the optimum procedure. Since n-dodecane diluent will probably be used when LMFBR fuels are processed, presumably the more simple of the solvent cleanup procedures will be suitable. In a batch cyclic demonstration simulating the processing of LMFBR fuels with 15% TBP--n-dodecane at ORNL, a two-stage wash with sodium carbonate solution appeared adequate.[24] Of the fission products, only ^{131}I accumulated in the solvent with cycling. The carbonate wash decreased the iodine concentration in the solvent entering the system by 20 to 35%.

4.7.3 Alternative Extractants

Over the past 25 years, many extractants of various types and structures have been developed for use in the recovery of metals. Of those which extract plutonium and uranium from nitrate solutions, only the phosphonates, the long-chain alkylamines, and certain organophosphorus esters similar to TBP appear worthy of serious consideration as alternatives to TBP.

The phosphonates [e.g., dibutylbutylphosphonate (DBBP)] have extraction properties similar to those of TBP but extract uranium and plutonium more strongly at equivalent acid concentrations.[34,35] Although this is an advantage in the extraction cycle, it is a disadvantage in the stripping cycle. In addition, DBBP extracts fission products more strongly than TBP, making it difficult to obtain decontamination factors equivalent to those obtained with TBP. At present there appears to be no significant advantage in substituting a phosphonate for TBP.

The long-chain alkylamines extract both uranium and plutonium from nitric acid; the extraction power varies over a wide range, depending on the amine type and structure.[5] The secondary, tertiary, and quaternary amines have adequate Pu(IV) extraction power for efficient plutonium recovery, but only the quaternary amines, and possibly the tertiary amines, have adequate uranium extraction power to allow coextraction of uranium and plutonium from nitric acid. Presumably, an amine process for co-recovery of plutonium and uranium could be developed; however, based on present knowledge, it is not apparent that the process would be better than (if as good as) the Purex process. A major disadvantage of the amines is that they provide relatively poor decontamination from ruthenium, which will be present in large amounts in irradiated LMFBR fuels. Amine extraction has been studied extensively,[36-38] and is presently used in France instead of ion exchange for the final purification of plutonium. It appears to be much more attractive for this use (see Sect. 4.8) than for first-cycle application.

Use of a phosphate ester with longer alkyl chains than TBP (e.g., trihexylphosphate) may have some important process advantages. These compounds have almost the same extraction properties as TBP but much lower distributions to the aqueous phase and less tendency toward third-phase formation. The former is particularly important since it should greatly decrease the tendency for precipitate (crud) formation in the solvent extraction system and in evaporators and result in the preparation of a final product solution that is contaminated with only a small amount of the organic phosphate extractant.

4.7.4 Contacting Equipment

Three main types of contactors have been used successfully in pilot and production operations of the Purex process. Pulsed columns (Fig. 4.10) have been utilized most extensively. In the United States, they are currently in production use at the Hanford and Idaho facilities and at the NFS plant, and will be used in the new General Electric plant.

Mixer-settlers were used for several years at the Savannah River plant and performed creditably. Recently, they were replaced by centrifugal contactors (Fig. 4.11), which have also performed satisfactorily. The latter type of contactor is classified as a "fast contactor," since the residence times of the phases in the contactor are very short and the solvent radiation exposure is correspondingly low. Another fast contactor, the stacked-clone contactor, (Fig. 4.12) is presently being developed at ORNL. Although it shows considerable promise, it has not yet been tested with radioactive feeds.

As compared with pulsed columns and mixer-settlers, fast contactors have the advantages of lower solvent radiation exposure, smaller size, smaller solvent inventory (reduced fire hazard), better control of criticality, and more rapid adjustment to steady-state conditions after start-up flow adjustment.

Pulsed Columns. - Pulsed columns have been the subject of much study.[39,40] Groenier *et al.*[41] correlated operating data from many sources, including some that were obtained from radiochemical operations. These data contained no documented evidence that the radiation field had a significant effect on column performance. However, the radiation power density in the columns was relatively low, and adverse effects could conceivably show up in the treatment of short-cooled LMFBR fuels since, in this case, the radiation dose to the solvent would be higher. Their report includes an extensive bibliography of documents describing the construction features, operational stability, modes of control, and operating data for pulsed columns. Scale-up of these columns is dependably predicted on the basis of the linear flow of solution (organic plus aqueous) through the column.

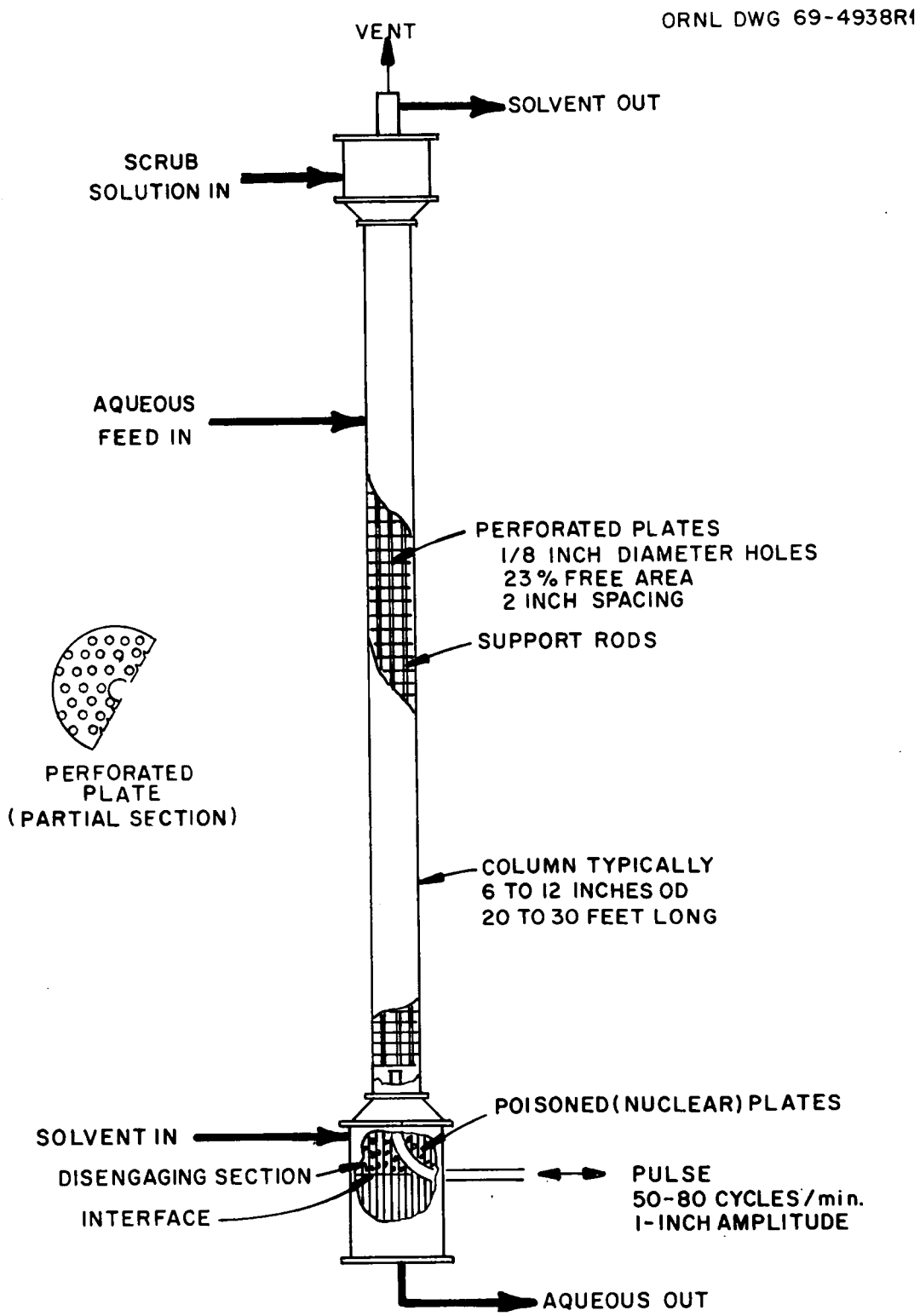

Fig. 4.10. Pulsed Column Designed for Organic-Phase-Continuous Operation.

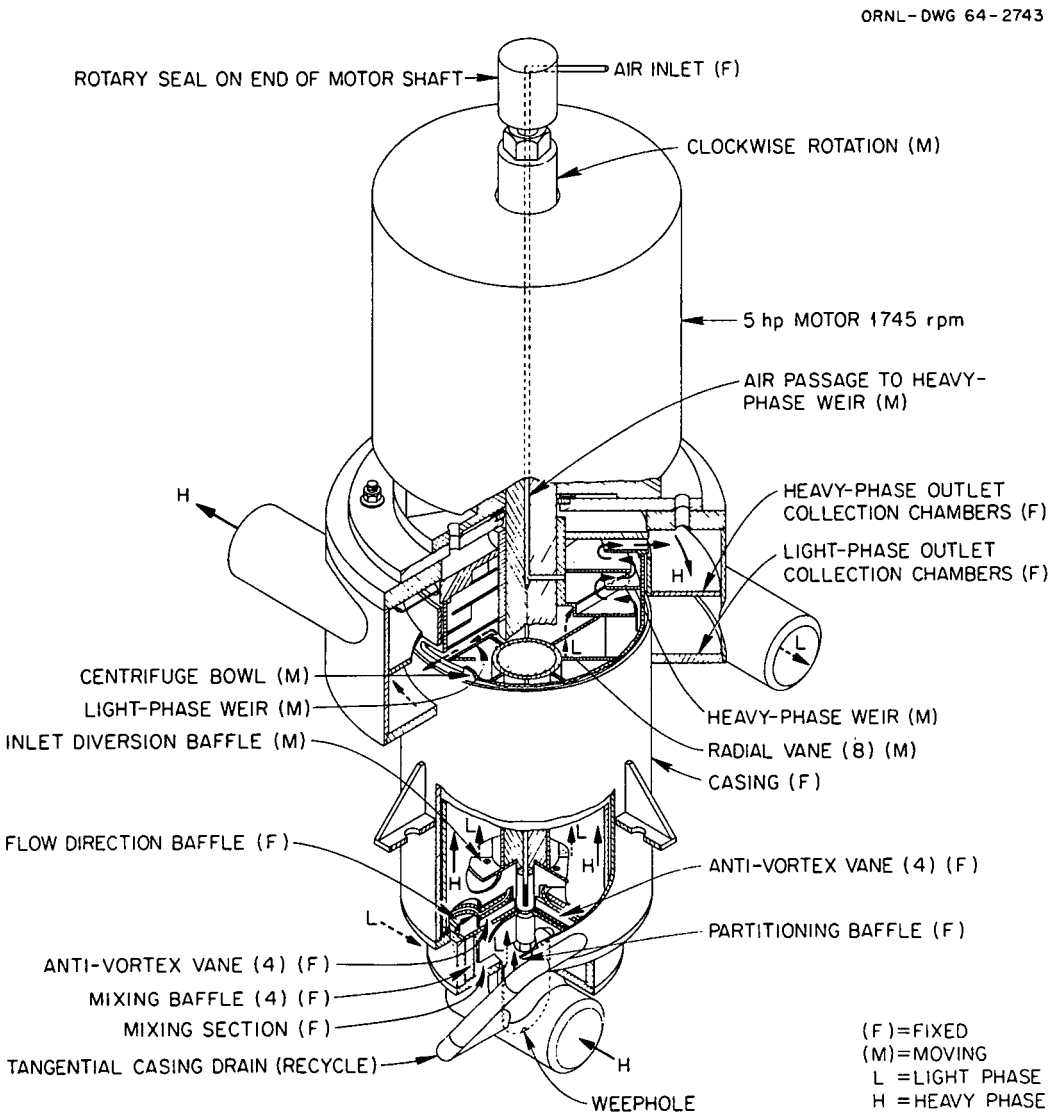

Fig. 4.11. Savannah River Centrifugal Contactor.

ORNL DWG 69-4942

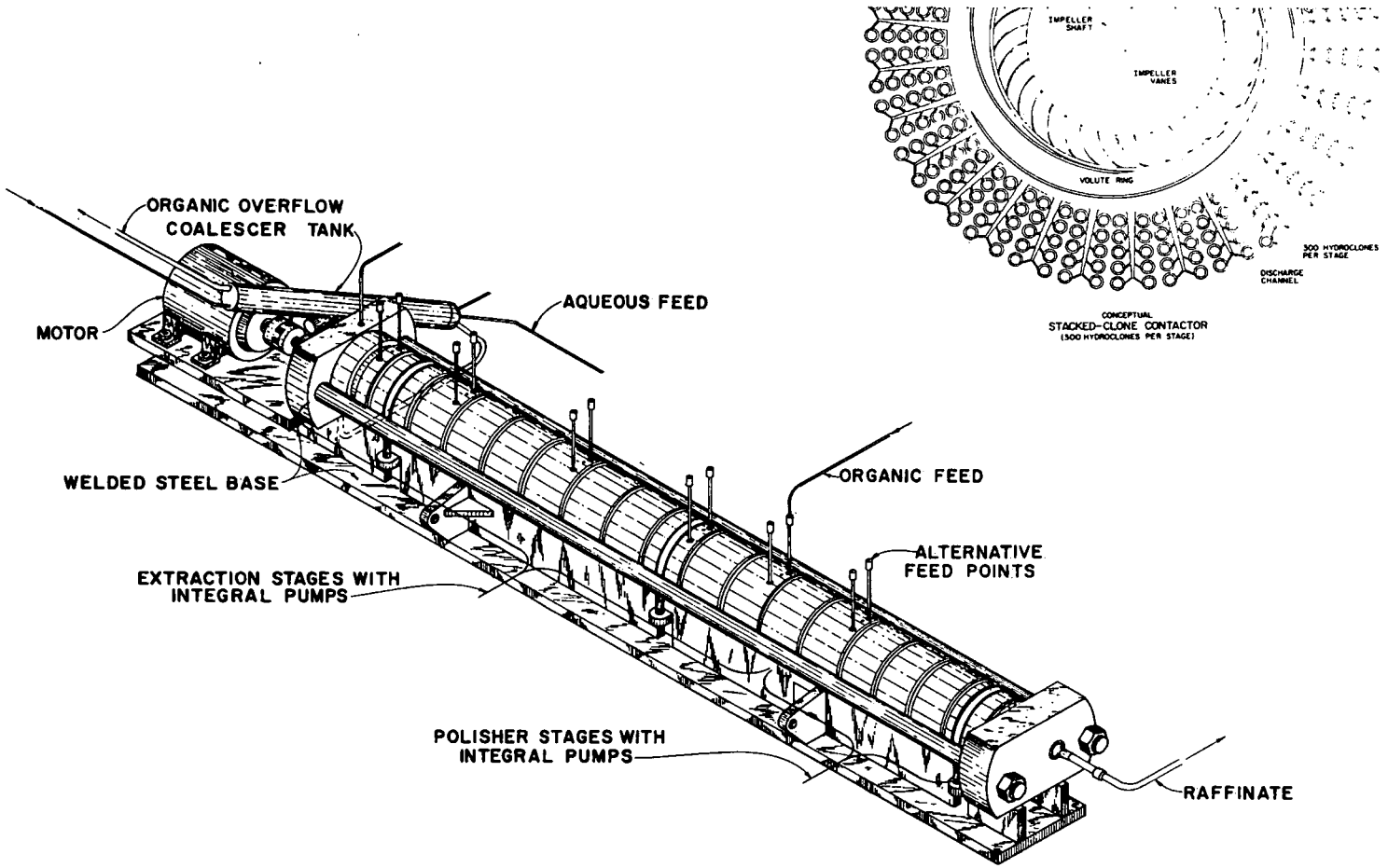

Fig. 4.12. ORNL Stacked-Clone Contactor.

Pulsed columns can be operated with a bottom interface (continuous organic phase). This is of considerable advantage from the standpoint of decontamination efficiency since any interfacial "cruds" that are formed are accumulated at the raffinate rather than at the "pure" end of the system. A disadvantage of this type of contactor is that it must be housed in a large (tall) shielded facility that has a relatively high capital cost.

Centrifugal Contactors. - The Savannah River centrifugal contactor has operated satisfactorily on a production basis for over a year.[42,43] It consists of 18 stages in three groups of six each. Each stage has a 10-in.-diam centrifuge bowl. Flow is cocurrent within a stage, but countercurrent between stages. Each stage has its own drive, mechanical seal, and interface control. The volume of each stage is about 4 gal (1 gal in the mixer and 3 gal in the settler). Mass transfer stage efficiencies are about 95%. The contactor operates in the organic-continuous mode.

A second extractor of the centrifugal type has recently been developed in France by Robatel and Mulatier.[44] It is estimated that radiation exposures, cost, and size requirements for this equipment would be comparable to those for the Savannah River contactor. An important advantage of the Robatel extractor is the incorporation of up to 12 stages in one machine utilizing a single drive mechanism and a single external grease-lubricated bearing.

The Stacked-Clone Contactor. - The stacked-clone contactor represents a rather new concept in solvent extraction equipment.[45] This device uses hydroclones to create both mixing and partial phase separation in each stage. The contactor has undergone extensive "cold" testing on a great variety of systems. The dependability and ease of operation are well-established, and radioactive operations do not appear to pose any significant new problems. Instrumentation is of a standard type; there is only one controlled interface position.

Scale-up studies show that the best scale-up method involves the manifolding of many small hydroclones in each stage (Fig. 4.12). Various details of the scale-up of stacked-clone contactors remain to be worked out. For instance, several contactors operating in parallel, each with a moderate

number of hydroclones per stage, might be preferred to a single machine with a very large number of hydroclones in each stage. The multiple contactor concept has the potential advantages of being more versatile in handling a varying plant throughput, and allowing a continuity of operation in case of component failure. Each stage of each contactor will have only one pump to drive the hydroclones. A 9-clone-per-stage, 11-stage device has recently been fabricated; the test results of this contactor should clarify many of the questions on scale-up.

4.7.5 Contactor Sizes and Capacities

Flow capacities (at flooding) in both aqueous continuous and organic continuous modes of operation have been predicted[46] for pulsed columns for two versions of the first-cycle Purex flowsheet, using the methods and equations of ref. 29. For aqueous continuous operation (assuming metal sieve plates having 1/8-in.-diam holes, a 23% free area, and 2-in. spacing), a total flow capacity of 111 ft/hr is predicted for a dilute flowsheet employing 15% TBP in a hydrocarbon diluent and an aqueous feed with about 70 g of heavy metal (10% plutonium) per liter. A flow capacity of 88 ft/hr is predicted for a 30% TBP flowsheet in which the aqueous feed has a heavy metal (10% plutonium) concentration of 200 g (U + Pu)/liter. For organic continuous operation, assuming metal nozzle plates of the same size and spacing as specified above, a total flow capacity of 335 ft/hr is predicted for the dilute flowsheet. The flow capacity for the flowsheet with 30% TBP and a 200-g (U + Pu)/liter feed is estimated at 312 ft/hr. All these capacities are for operation at 40°C and for a pulse having a 1-in. amplitude and a frequency of about 50 cpm. Usually, the organic continuous mode of operation is preferred because any accumulation of debris at the interface may be flushed out with the raffinate. The much higher capacity (about a factor of 3) obtainable using this mode of operation is a very important added advantage and is primarily due to the faster coalescence rates of organic-continuous dispersions.

For operation at about 80% of the predicted flow capacity (i.e., 80% of flooding) with the organic phase continuous, the extraction-scrub section of the first cycle of a 5-metric-ton-per-day LMFBR plant would require two

12-in.-diam columns or their equivalent. For the more-concentrated flowsheet, only one 12-in.-diam column would be needed. Assuming that 11 extraction plus scrub stages are used and that the height of each stage would be about 2.5 ft, the columns would be 27.5 ft high, exclusive of end sections.

Pulsed columns for the partitioning and uranium stripping operations in the first cycle would be at least as large in diameter as those for extraction. It is estimated that two 12-in.-diam partitioning columns and two 15-in.-diam stripping columns would be required.

The above estimates of equipment size do not take criticality into consideration. This will be a controlling factor which will dictate the size and number of columns needed, as well as the advantages or disadvantages of a dilute vs a concentrated flowsheet. Criticality guidelines that will greatly aid in the development of preliminary equipment designs have recently been published.[47] It appears that a critical diameter as large as 15 in. applies to mixed plutonium-uranium nitrate solutions that contain 300 g of heavy metal (20% Pu) per liter and a soluble poison (0.2 M Cd) and are reflected by 3 cm of stainless steel. However, if the plutonium is concentrated by excessive reflux or if it segregates through formation of a second organic phase, critical diameters are much lower. Extensive testing under a variety of assumed accident conditions is needed to define the critically safe equipment sizes for different TBP concentrations and aqueous-feed plutonium concentrations. The critical column diameter can be appreciably increased by using poisoned (0.2% Gd) stainless steel pulse plates. Also, an alternative to the use of smaller columns may be an annular column design with a poisoned central core.

Centrifugal Contactors. - It is estimated that a centrifugal contactor of the type and stage size used at the Savannah River Plant could handle the flow (~ 42 gal of organic plus aqueous per minute) in a 5-metric-ton-per-day-plant using the 15% TBP flowsheet and operating at about 70% of flooding. A 12-stage unit, which would provide about 11 ideal stages for extraction-scrubbing, would occupy a space about 10 x 10 x 7 ft high. Centrifugal extractors and other short-residence contactors may not be

suitable for use in the partitioning cycle since the plutonium reduction reaction may be too slow; however, they could be used for uranium stripping.

With respect to the French centrifugal contactor, it is estimated that three Robatel units, each 67 cm in diameter and having 12 stages, would be needed for the first-cycle extraction-scrubbing operation in a 5-metric-ton-per-day plant.

Stacked-Clone Contactor. - A 5-metric-ton-per-day LMFBR fuel processing plant would require a considerably larger stacked-clone contactor for the first-cycle extraction-scrubbing operation than any of the models that have been tested. A recent estimate[48] indicates that a single stacked-clone contactor unit containing 28 stages (11 ideal stages), each with two hundred seventy-two 0.9-in.-diam hydroclones, would handle the flows (15% TBP flowsheet) for such a plant. The contactor would occupy a space about 2 ft in diameter by 8 ft long. A contactor stage having 300 hydroclones of this size has been designed. Fabrication techniques for a contactor of this design would likely be quite different from those used in the past. Automated methods for producing the many parts would become feasible and offer substantial reductions in the cost per hydroclone.

4.7.6 Solvent Radiation Exposures in Different Contactors

Solvent radiation exposures[46,48] were calculated at ORNL for the processing of 30-day-cooled mixed core and blanket fuel (33,000 Mwd/metric ton burnup). Sample calculations are shown in refs. 7 and 46. The exposures in pulsed column cartridges and end sections were predicted by using the equations of ref. 21. It was assumed that all of the beta energy was absorbed by the organic and aqueous phases; estimates of the amount of absorbed gamma radiation were based on gamma radiation attenuation considerations obtained from ref. 49. In the case of pulsed columns, the alpha radiation dose and the dose due to accumulated radioiodine in the solvent were each assumed to be 10% of the beta-gamma dose.

Solvent exposures were estimated to be in the range of 0.02 to 0.03 whr liter^{-1} cycle^{-1} for both the Savannah River centrifugal contactor and the stacked-clone contactor when a dilute (15% TBP, 70 g heavy metals/liter)

flowsheet is used. These estimates include an allowance of 0.01 whr liter^{-1} cycle^{-1} to cover the combined doses due to alpha radiation and to accumulated radioiodine in the solvent. The solvent exposures in these contactors would be far below the level at which deleterious effects from solvent irradiation have been found.

Solvent exposures in pulsed columns were estimated assuming operation at 80% of flooding. For organic continuous operation, exposures were estimated at 0.10 and 0.27 whr liter^{-1} cycle^{-1}, respectively, for the dilute (see above) and the concentrated (30% TBP, 200 g heavy metals/liter) flowsheets.

The solvent exposure estimates are considered to be conservative, that is, on the high side, for the following reasons:

1. The radiation density was assumed to be the same in both the extraction and scrub sections, whereas, in actual operations, only a small fraction of the fission products would reach the scrub section.
2. A uniform radiation field (i.e., an infinitely small aqueous drop size) in the column was assumed. Actually, the aqueous drop sizes in pulsed columns would be sufficiently large that a significant fraction of the beta energy would be absorbed within the aqueous drops.
3. It was assumed that all of the fission products would be present in the feed. Dissolution tests, however, have shown that significant fractions of some of the more important fission products (e.g., ruthenium, zirconium-niobium) would be in the leach residue, which probably would be separated from the aqueous feed prior to solvent extraction.

The following factors could lead to higher doses than those calculated:

1. A decrease in throughput capacity of the contactor below the assumed values would result in a corresponding increase in the solvent radiation dose. This could apply particularly if closely

spaced poisoned plates are used in the pulsed columns for criticality control.

2. No allowance was made for possible accumulation of highly radioactive cruds in the equipment. Although this is not expected in normal operation, it could occur during periods of maloperation.

4.8 Plutonium Purification - Task 8

This task covers the process steps for purifying plutonium, starting with the plutonium product solution from the first Purex cycle and continuing through the preparation of a product solution of adequate purity and concentration for delivery to the fuel refabrication operation.

Plutonium chemistry is reasonably well known, although the high plutonium content of LMFBR fuels makes some process revisions desirable. The large amounts of plutonium to be handled provide a strong incentive for improving the efficiency of the plutonium purification processes since the cost of these operations will represent a much larger fraction of the total processing costs than previously. In particular, there appears to be a need for a higher-capacity process than ion exchange for final decontamination of the plutonium.

4.8.1 Second TBP Cycle

The first-cycle plutonium product is usually purified in a second TBP cycle,[1-3] and the product from this cycle is finally purified by ion exchange. Prior to the second TBP cycle, the Pu(III) is oxidized to Pu(IV) by adding sodium nitrite and heating (or by sparging with NO gas), and the nitric acid concentration is adjusted to 3 to 4 M. After extraction, the plutonium is stripped from the solvent phase with dilute nitric acid; usually some sulfuric acid is added to improve the stripping efficiency. A small amount of a plutonium reductant such as Fe(II) or U(IV) can also be added to the dilute acid to increase the concentration of the plutonium in the product solution and to ensure complete stripping of plutonium from the solvent.

A second-cycle flowsheet using 15% TBP was successfully demonstrated at ORNL in batch countercurrent tests with simulated second-cycle LMFBR feeds.[4] More than 99.8% of the plutonium was recovered in the extraction cycle. The solvent was stripped with 0.15 M HNO_3--0.02 M $Fe(NO_3)_2$--0.01 M hydrazine solution to give a product solution containing more than 40 g of plutonium per liter.

4.8.2 Ion Exchange Processing

In the past, purification of the plutonium has been accomplished by cation exchange; however, present practice favors the use of anion exchange resins.[2-7]

The cation exchange resins (e.g., Dowex 50) sorb Pu(III) effectively. The process, therefore, can handle plutonium product solutions directly from the reduction-partitioning column, without adjustment of the plutonium valence state.[7] Following the sorption cycle, the resin is washed with dilute sulfuric acid to preferentially elute uranium and certain fission products. The plutonium is then eluted with fairly concentrated nitric acid containing a holding reductant to prevent oxidation of Pu(III) to Pu(IV). A principal objective of the use of the cation exchange process is to concentrate the plutonium. This would not be an advantage for LMFBR fuels since the plutonium in these fuels is already relatively concentrated. The process provides little separation from ruthenium (DF of 2 to 3) and only moderate separation (DF ~10) from zirconium-niobium. The sulfuric acid wash step is also a disadvantage. Therefore, cation exchange processing has no apparent merit for use with LMFBR fuels.

Anion Exchange Processing. - Plutonium(IV) is sorbed more strongly than the other valence states by anion exchange resins (e.g., Dowex 1, Permutit SK). Maximum sorption efficiency occurs when the nitrate concentration is about 7 M. The plutonium is usually eluted with dilute nitric acid, but it can also be eluted by reduction to the trivalent state with hydroxylamine nitrate. The rates of both sorption and elution are slow. Practicable exchange rates are obtained by operating at 60 to 70°C.

Chemical and Radiation Stability. - A few instances of pressurization and fire have occurred in plutonium processing facilities employing anion exchange resins.[5,8] The reasons for these occurrences are not thoroughly understood; however, in some cases, the autocatalytic buildup of nitrite was thought to be a contributing factor. Chemical attack is rapidly accelerated at temperatures and pressures that are above normal. The radiolytic decomposition of anion exchange resins has been studied extensively. The physical effects of radiation damage are large volume changes, softened beads, and increased pressure drop. The resins show a rapid loss of capacity at radiation doses above 10^8 rads. Estimates of resin life expectancy during the continuous processing of ^{239}Pu with Permutit SK, the most stable of the available resins, range from two to four months.[7] This range might be appreciably reduced when LMFBR fuels are processed, owing to the increased alpha activity of ^{238}Pu and to the higher concentrations of plutonium that would be present in the system.

Equipment. - Both static-bed and continuous anion exchange methods have been used on a production basis. With the continuous units, loss of plutonium to the raffinate streams is relatively high (2 to 5%), requiring that these streams be recycled. Recycle is accomplished simply by evaporative concentration and return of the concentrate to the feed stream of the second-plutonium-cycle solvent extraction column. The larger amounts of plutonium present in fast breeder fuels will demand higher-capacity ion exchange equipment than is currently used, at least for large plants. For a processing facility of 5-metric-tons (U + Pu)-per-day capacity, approximately 15 continuous ion exchange systems (8 in. in diameter, as limited by criticality) operating in parallel would be required. It is obvious that the presence of such a large number of parallel systems is undesirable. For a 1-metric-ton-per-day plant, this disadvantage would, of course, be less limiting, and the three to four parallel units needed can be considered to be more practical.

It should be possible to circumvent limitations on column diameter imposed by criticality considerations and to greatly increase the capacity of a static-bed column by use of an annular column design (with a poisoned central core). The use of extremely fine resin beads in high-pressure

static-bed columns, a system that has been used in the TRU program to separate higher actinides,[9] appears also to be worthy of consideration since the usual unfavorable absorption-elution kinetics, which are controlled by diffusion, should be greatly improved in this system.

4.8.3 Amine Extraction of Plutonium

The final purification of plutonium by amine extraction, rather than by anion exchange resin sorption, has been studied extensively and is presently used on a production basis in France.[10-12] The amines extract plutonium rapidly at ambient temperatures; the extraction coefficients reach a maximum when the nitrate concentration is about 7 M. Almost all of the data have been obtained with tertiary amines, which cannot be stripped effectively with dilute nitric acid but require use of complexing agents such as sulfate or acetic acid. A stripping procedure that involves precipitating plutonium directly from the solvent phase as the oxalate has also been studied.

Secondary amines, which are weaker extractants and can be stripped with dilute nitric acid, appear promising.[12,13] A flowsheet for the extraction of Pu(IV) from 5 M HNO_3 with di(tridecyl)amine in diethylbenzene has been demonstrated in a batch countercurrent system.[13] The extract was scrubbed with 4 M HNO_3 and stripped with 0.15 M HNO_3 to give a product solution containing about 50 g of plutonium per liter. This solution should be sufficiently concentrated (in plutonium) to be fed directly to a fuel refabrication process (e.g., a sol-gel process or precipitation processes) for reconstituting the fuel. Additional testing of the secondary amines, particularly with respect to the separations obtainable from fission products, is needed to determine if they are competitive with tertiary amines.

Amine extraction appears to have considerable advantage over anion exchange for final purification of the plutonium in large plants. The more favorable kinetics of the amine process, as compared with the resin process, allow higher throughputs in this system. High-capacity amine extraction equipment can be constructed by taking advantage of the inherent high capacity of solvent extraction equipment and of the capability of installing fixed poisons in the contacting devices. Thus, criticality restrictions

will demand fewer parallel lines of equipment (and possibly only a single line) for amine extraction than for ion exchange.

Amine Stability. - The radiolytic degradation products of amines are relatively innocuous, and large radiation doses can be absorbed without any noticeable effect on the process performance. The amines are subject to rapid chemical degradation in the presence of high concentrations of nitrous acid. However, this does not appear to be a significant problem since the feed solution to the amine process can be treated to destroy nitrite.

4.8.4 Adaptation to LMFBR Fuels

The processes used for final purification of plutonium from thermal fuels can undoubtedly be adapted to LMFBR fuels. However, the large amounts of plutonium in the latter fuels will require some adjustments of flowsheet conditions and will increase criticality problems. The relatively high alpha activity of the LMFBR fuels (0.5% ^{238}Pu) will generate neutrons by (α, n) reaction on oxygen and may require that the plutonium be refabricated in a shielded facility. In this case, the required decontamination factors from fission products would not need to be as high as in the past; consequently, there is a potential opportunity for decreasing the number of plutonium purification steps. With respect to the overall purification scheme, there appears to be considerable potential for improving process economics by simplifying and integrating the purification cycles.

4.9 Waste Management - Task 9

This section summarizes the status of the technology for management of radioactive wastes produced by the aqueous method for processing nuclear fuels. Both the technologies used at present and recent developments are described. The term "waste management" includes treatment, storage (which implies retrievability), and disposal (which implies the reverse). Estimates of the magnitude of the problem and the cost of waste management are also presented. Particular emphasis is placed on comparing waste management

problems for Liquid Metal Fast Breeder Reactor (LMFBR) fuels with present-day or advanced Light Water Reactor (LWR) fuel. The management problems for aqueous and gaseous wastes from LMFBR fuels are quite similar to those for LWR fuels. However, the presence of higher concentrations of short-lived fission products in LMFBR wastes adds a further complication.

The implications of probable new waste management concepts are discussed. For example, nuclear installations not located on a government area may be required to ship all radioactive residues to a government-controlled disposal, or burial, ground. Also, only high- and low-level wastes would be considered on a regulatory basis for treatment and disposal. Intermediate-level wastes would be transitory in an installation and would thus be converted to, or combined with, high- or low-level wastes, based on their content of long-lived alpha-emitting nuclides as well as on their beta and gamma nuclide content.

4.9.1 Summary and Conclusions

1. The methods currently used for waste management, supplemented by technology in an advanced state of development, should be satisfactory for handling the wastes that accumulate from the aqueous processing of LMFBR fuels. More-advanced technology, which is now in the early stages of development, may decrease costs and increase safety of waste management.

2. The aqueous high-level wastes from processing LMFBR fuels will not differ significantly from advanced LWR wastes with regard to ionic content except for the presence of stainless steel (rather than Zircaloy) from partial dissolution of cladding and, perhaps boron, which is added during processing as a soluble poison. The heat generation rate per ton of processed fuel will be higher because of the shorter cooling times (~30 days vs 150 days) and higher reactor power ratings (~90 to 180 kw/kg vs 15 to 40 kw/kg). The heat generation rates for LMFBR and LWR wastes will be about the same after approximately 3 years of cooling for comparable burnups.

3. High-level liquid wastes can be stored safely in tanks that have been provided with adequately engineered safety features. Because of the

requirement for continuous removal of heat, the effectiveness of the containment system depends on a very high degree of surveillance.

4. The only current, technically acceptable alternative to tank storage of high-level wastes with high heat generation rates is immediate solidification of the wastes. Currently, the disposal of the solidified wastes by emplacement in bedded salt deposits is believed to be the safest method and has been shown to be technologically feasible. Alternatively, disposal of high- and intermediate-level solidified wastes of low specific heat-generation rates by hydrofracturing or emplacement in bed-rock caverns may be acceptable at government-owned sites with suitable geological characteristics.

5. The volumes of liquid and solid high-level wastes from LWR and LMFBR fuels to be produced by the year 2000 are not excessive, that is, 60 million gal of liquid, or 600,000 ft^3 of solids, as compared with the more than 80 million gal of liquid waste now stored in tanks in the United States.

6. Four waste solidification processes have been developed in the United States and have been successfully demonstrated on an industrial scale with full levels of radioactivity for current LWR fuels (up to 20,000 Mwd/metric ton, cooled 250 to 320 days). The heat generation rate is the principal limiting factor for these processes. Process modifications that would produce solids with higher thermal conductivities are being developed. The objective of this program is to make feasible the early solidification of wastes from LMFBR fuels that have been cooled 30 days.

7. Our present assessment is that solidified high-level wastes can be shipped safely and economically. The shipment of liquid wastes is not considered safe because of considerations of steam-pressure buildup within casks following a loss-of-coolant incident. The shipment of solidified wastes is inherently simpler than the shipment of spent fuels because of the lower heat load and the absence of gaseous fission products.

8. The feasibility of disposing of solidified waste in natural salt formations has been demonstrated in a salt mine in Kansas, using spent reactor fuel instead of actual wastes. Satisfactory results were obtained

with regard to the safety of handling radioactive materials underground, the stability of salt to heat and radiation, and the creep and plastic flow of the salt. These results should make possible the design of a safe disposal facility.

9. The estimated cost for the complete management of solidified wastes from LWR fuel processing, including interim liquid storage, solidification, interim solid storage, 1000-mile shipment, and permanent disposal in bedded salt is about 0.02 mill/kwhr(electrical), based on government financing, 4% interest, and 1962 costs. The cost would be about the same if the wastes were stored indefinitely in tanks. Revised estimates reflecting private financing are being prepared which take into account taxes, a return on investment, and current costs and interest rates. The revised total cost of waste management is expected to range from 0.03 to 0.05 mill/kwhr(electrical). Costs are not expected to be significantly different for managing wastes from LMFBR fuel processing.

10. Large inventories of radioactive gases such as Kr, I_2, $^3H_2$, and Xe will be collected in future fuel processing plants to prevent their discharge to the atmosphere. The present practice involves eliminating most of the short-lived nuclides by decay prior to processing. Additional decontamination of the off-gases is achieved using caustic scrubbers, silver nitrate towers, or charcoal beds when necessary. Improved techniques for the separation and containment of these gases are being considered to prevent accidental release of large amounts of radioactive gases.

11. Intermediate-level wastes (i.e., liquid or solid wastes which contain modest levels of beta and gamma nuclides) will be considered transitory; those requiring containment because of their long-lived actinide content will probably be combined for processing with the high-level wastes. Initial estimates indicate that this procedure should not increase the cost of waste management significantly. A loss of 0.1% of the plutonium to the liquid or solid wastes in a plant processing 5 metric tons of 20% PuO_2--80% UO_2 per day would amount to 365 kg of plutonium per year.

12. Low- and intermediate-level wastes that do not contain significant amounts of long-lived actinides can be handled by techniques similar

to those presently being used in industry. Liquid wastes are generally treated by evaporation, flocculation, or ion exchange, and the residues are stored or buried with other miscellaneous solid wastes. Techniques have been developed for dispersing intermediate-level waste residues in asphalt or polyethylene (to decrease their solubility prior to burial) and for injecting waste-cement slurries into deep geologic formations by hydro-fracturing.

13. The volumes of solid waste, from the metallic components of LWR and LMFBR fuel elements and other miscellaneous waste, that will be produced by the year 2000 are not expected to be excessive, that is, 1 million and 49 million ft^3, respectively.

4.9.2 Magnitude of the Problem

High-Level Aqueous and Gaseous Wastes. - Estimates have been made of the amounts of fuel to be processed and the resulting waste volumes to be accumulated by the year 2020.[1,2] The reactors considered are light-water and fast breeders.

For waste projections, it is assumed that LWR fuel is exposed to 33,000 Mwd/metric ton at an average specific power of 30 Mw/metric ton. The amount of fuel of this assumed exposure level to be processed annually will increase from 52 tons in 1970 to 4800 tons in 2000 (comparison of Tables 4.4 and 4.5). Assuming that the fuel is processed by aqueous methods and that the resulting waste is concentrated to 100 gal per 10,000 Mwd(thermal), the annual production of waste will increase from 17,000 gal in 1970 to 1.6 million gal in 2000. If it is stored as a liquid, 39 million gal will accumulate by the year 2000. If it is converted to a solid form, waste volumes may be reduced by a factor of about 13.

Wastes from LMFBR's are expected to become significant in the 1980's (Table 4.4). The quantity of fuel and blanket to be processed annually, assuming an average exposure of 33,000 Mwd/metric ton, will increase from 360 metric tons in 1985 to 9230 metric tons in the year 2000. It is estimated that the aqueous processing of this material will produce 21 million gal of accumulated liquid waste, concentrated to a volume of 100

Table 4.4. Projected Wastes from LMFBR Reactors (Aqueous Processing of All Fuels)

	Calendar Year			
	1985	1990	2000	2020
Installed capacity, 10^3 Mw(electrical)[a]	28	145	546	1669
Fuel processed, 10^3 metric tons/year[b]	0.36	2.15	9.23	27.6
Volume of waste generated, as liquid[c]				
Annually, 10^6 gal/year	0.118	0.71	3.02	9.08
Accumulated, 10^6 gal	0.248	2.4	20.9	150.6
Volume of waste generated, as solid[d]				
Annually, 10^3 ft^3/year	1.18	7.1	30.2	90.8
Accumulated, 10^3 ft^3	2.48	24	209	1504
Accumulated radioisotopes[e]				
Total weight, metric tons	25	260	2200	15,640
Total activity, megacuries	4,388	30,000	146,450	523,300
Total heat-generation rate, Mw	17.4	117	563	1949
^{90}Sr, megacuries	31.8	300	2465	15,500
^{137}Cs, megacuries	78.3	740	6,070	38,600
^{129}I, curies	39.1	380	3300	22,690
^{85}Kr, megacuries	7.2	66	489	2620
^{3}H, megacuries	0.653	6.0	46.1	252
^{238}Pu, megacuries[f]	0.18	1.98	9.1	141.5
^{239}Pu, megacuries[f]	0.013	0.128	1.114	8.01
^{240}Pu, megacuries[f]	0.0161	0.156	1.38	10.0
^{241}Pu, megacuries[f]	2.12	19.5	150.7	835
^{242}Pu, curies[f]	48	469	4063	29,094
^{241}Am, megacuries	1.18	11.4	100	716
^{243}Am, megacuries	0.037	0.36	3.12	22.4

Table 4.4. (Cont'd.)

	Calendar Year			
	1985	1990	2000	2020
^{244}Cm, megacuries	0.73	7	55	321
^{242}Cm, megacuries	14.5	95	415	1279

[a]Data from Phase 3, Case 42, Systems Analysis Task Force (April 11, 1968).

[b]Based on an average exposure of 33,000 Mwd/metric ton, and a delay of 2 years between power generation and fuel processing.

[c]Assumes wastes concentrated to 100 gal per 10^4 Mwd(thermal) and a delay of 2 years between power generation and waste generation.

[d]Assumes 1 ft^3 of solidified waste per 10^4 Mwd(thermal).

[e]Assumes core continuously irradiated at 148 Mw/metric ton to 80,000 Mwd/metric ton, axial blanket to 2500 Mwd/metric ton at 4.6 Mw/metric ton, radial blanket to 8100 Mwd/metric ton at 8.4 Mw/metric ton, and fuel processing 30 days after discharge from reactor.

[f]Assumes 0.5% of Pu in spent fuel is lost to waste.

Table 4.5. Projected Wastes from U. S. Nuclear Power Economy (Aqueous Processing of All Fuels)

	Calendar Year				
	1970	1980	1990	2000	2020
Installed capacity, 10^3 Mw(electrical)[a]	14.4	153	368	735	2210
Fuel processed, 10^3 metric tons/year[b]	0.052	2.95	8.16	14.0	41.7
Volume of waste generated, as liquid[c]					
Annually, 10^6 gal/year	0.017	0.97	2.69	4.60	13.7
Accumulated, 10^6 gal	0.017	4.40	23.8	60.1	238
Volume of waste generated, as solid[d]					
Annually, 10^3 ft^3/year	0.17	9.73	26.9	46.0	137
Accumulated, 10^3 ft^3	0.17	44.0	238	601	2380
Accumulated radioisotopes[e]					
Total weight, metric tons	1.75	451	2440	6200	24,600
Total activity, megacuries	2.08	18,900	84,500	209,000	666,000
Total heat-generation rate, Mw	0.91	81.6	343	807	2520
^{90}Sr, megacuries	3.98	962	4640	9550	29,400
^{137}Cs, megacuries	5.27	1280	6540	15600	57,500
^{129}I, curies	1.85	476	2700	7550	32,200
^{85}Kr, megacuries	0.56	124	567	1190	3900
^{3}H, megacuries	0.033	7.29	36.2	89.5	332
^{238}Pu, megacuries[f]	0.002	1.20	8.28	30.7	166
^{239}Pu, megacuries[f]	0.00009	0.022	0.235	1.31	8.45
^{240}Pu, megacuries[f]	0.00013	0.0409	0.395	1.91	11.4
^{241}Pu, megacuries[f]	0.0295	6.63	47.2	191	909
^{242}Pu, curies[f]	0.354	91	910	4870	30,900

Table 4.5. (Cont'd.)

	Calendar Year				
	1970	1980	1990	2000	2020
^{241}Am, megacuries	0.0089	2.31	22.7	121	763
^{243}Am, megacuries	0.0009	0.232	1.49	5.19	27.0
^{244}Cm, megacuries	0.128	29.9	137	255	700
^{242}Cm, megacuries	0.725	43.2	185	487	1490

[a]Data from Phase 3, Case 42, Systems Analysis Task Force (April 11, 1968).

[b]Based on an average exposure of 33,000 Mwd/metric ton, and a delay of 2 years between power generation and fuel processing.

[c]Assumes wastes concentrated to 100 gal per 10^4 Mwd (thermal) and a delay of 2 years between power generation and waste generation.

[d]Assumes 1 ft^3 of solidified waste per 10^4 Mwd(thermal).

[e]Assumes LWR fuel continuously irradiated at 30 Mw/metric ton to 33,000 Mwd/metric ton, and fuel processing 90 days after discharge from reactor; LMFBR core continuously irradiated to 80,000 Mwd/metric ton at 148 Mw/metric ton, axial blanket to 2500 Mwd/metric ton at 4.6 Mw/metric ton, radial blanket to 8100 Mwd/metric ton at 8.4 Mw/metric ton, and fuel processing 30 days after discharge.

[f]Assumes 0.5% of Pu in spent fuel is lost to waste.

gal per 10,000 Mwd(thermal), by the year 2000. This waste, on conversion to solids, would yield a volume of 209,000 ft^3.

The total accumulation of wastes for all reactor types to the year 2020 is shown in Table 4.5. It was concluded that the volume of waste to be produced in the aqueous processing of all nuclear fuels by the year 2000 is not excessive (i.e., 60.1 million gal of liquid, or 600,000 ft^3 if converted to solids). This compares with the approximately 80 million gal of liquid waste now stored at the USAEC production sites.

The heat generation rates of wastes produced by the processing of LWR and LMFBR fuels have important process significance. The heat generation rate of the waste produced by processing one ton of spent LMFBR or LWR fuel is shown in Fig. 4.13 as a function of the time following discharge of the fuel from the reactor. The fuel irradiation conditions were 33,000 Mwd/metric ton at 24.5 Mw/metric ton for the LWR, as compared with 80,000 Mwd/metric ton at 89.3 Mw/metric ton for the LMFBR. The ratio of heat generation of these fuels on discharge from the reactors is equal to the ratio of the reactor operating powers (89.3/24.4 = 3.63); however, it decreases over the next 3 years to that of the respective fuel exposures (80,000/33,000 = 2.42), and remains at that value thereafter.

No estimate has been made of the volumes of gaseous wastes that would be produced in the fuel processing plants. However, calculations have been made of the megacuries of ^{3}H and ^{85}Kr that would be accumulated (Tables 4.4 and 4.5). In addition, in future plants, large inventories of short-lived gaseous fission products will be accumulated. For example, the equilibrium amounts of ^{131}I, ^{131}Xe, and ^{133}Xe in a plant processing 6 metric tons of 30-day-cooled Atomics International Reference Oxide LMFBR fuel (mixed core and blankets) per day would be as follows: ^{131}I, 1.04×10^7 curies; ^{131}Xe, 6.3×10^5 curies; and ^{133}Xe, 3.66×10^6 curies. In present commercial fuel processing plants, these short-lived radionuclides are eliminated by decay of the fuel prior to processing. The long-lived ^{3}H and ^{85}Kr are discharged to the environment. In future plants, both the short-lived gaseous nuclides, which are contained in short-cooled LMFBR fuels, and the long-lived nuclides must be retained.

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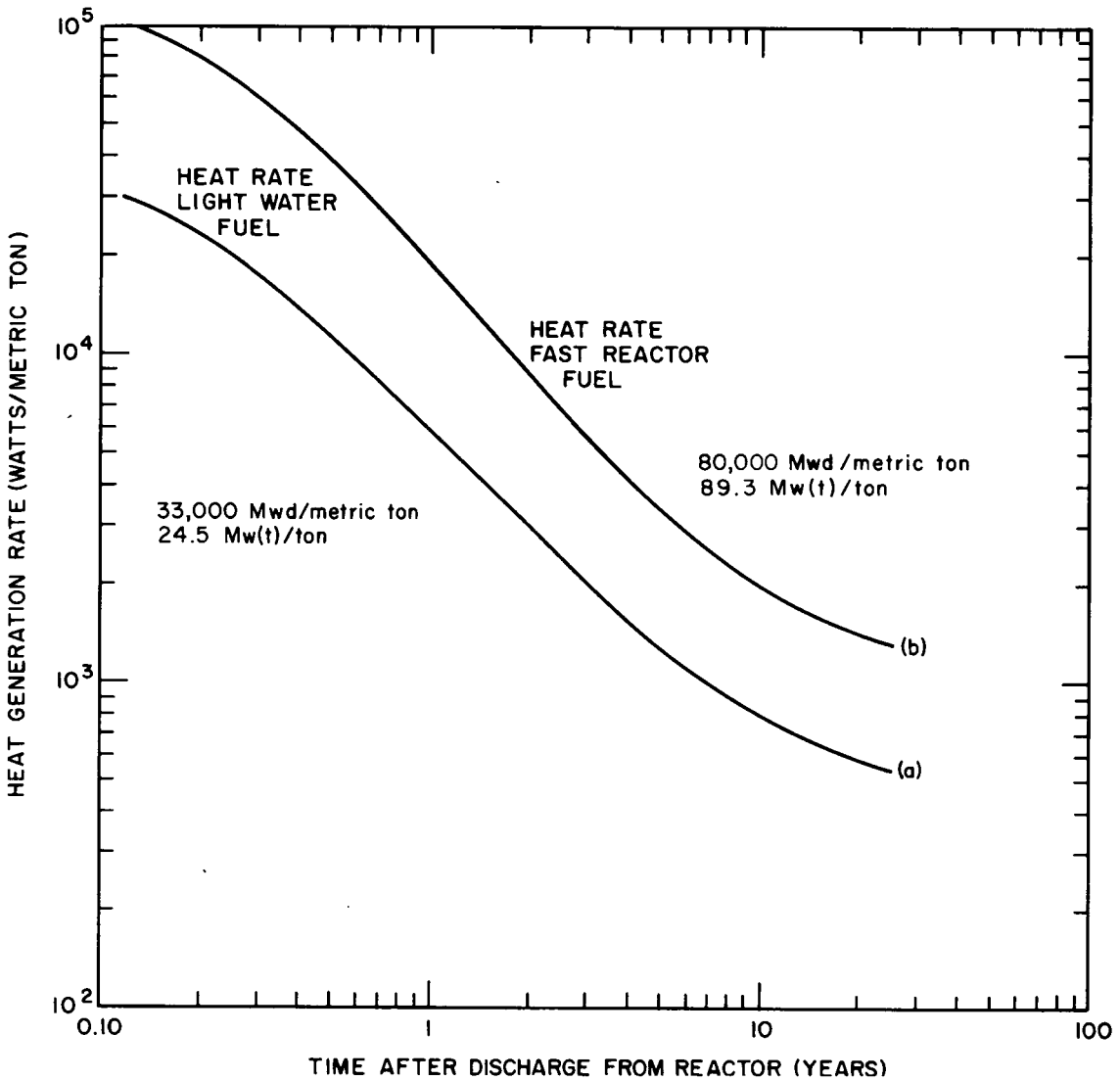

Fig. 4.13. Variation of Fission Product Heat Generation Rate (watts) with Decay Time for One Ton of Spent Fuel from (a) Light-Water Reactors and (b) Fast Breeder Reactors.

Low- and Intermediate-Level Liquid Wastes. - The estimated rates at which intermediate- and low-level liquid wastes will be produced by fuel processing are shown in Table 4.6. A large fraction of the intermediate-level waste will probably contain significant amounts of long-lived alpha nuclides and will be combined with the high-level wastes. In addition to these volumes, it can also be assumed that approximately 3 million gal of low-level waste will be produced annually at each processing plant from sources such as cell drainage, equipment decontamination, and laboratory sinks.

Solid Wastes. - If a mechanical decladding method, such as the shear-leach process, is used as the initial processing step, approximately 270 kg (2.1 ft^3) of metallic cladding will be produced per ton of fuel, based on experience with LWR fuels.[2] Table 4.7 shows projected annual and accumulated volumes of solid wastes produced by spent fuel processing from 1970 to the year 2020. A tentative waste management flowsheet for LMFBR fuel shows 650 kg of alpha-contaminated stainless steel cladding waste, and 700 kg of fuel element hardware containing induced activity, per metric ton of heavy metals.[3] The former would probably be considered high-level waste, and the latter low-level, for waste disposal purposes. Experiments have shown that the metallic wastes can be compressed to solid billets;[3] that is, 1350 kg of stainless steel cladding can be compressed to 8.7 ft^3 (a bulk density of 5.6 g/cc). The precise amount of plutonium lost to the cladding in the case of LMFBR fuel is unknown; however, a loss of 0.01% would amount to 36 kg of plutonium per year for a 5-metric-ton/day plant with fuel containing 20% PuO_2. Losses of about 0.1% have been observed with LWR fuels.

4.9.3 Status of Waste Management Technology

Chemical Composition of High-Level Waste. - Five waste compositions that cover the expected ranges of chemical compositions of wastes have been used in development studies.[4] The range in concentrations of the significant chemical constituents for the four LWR wastes is as follows (based on a volume of 100 gal/metric ton): Fe, 0.05 to 1.0 M; Cr, 0.01 to 0.02 M; Ni, 0.005 to 0.01 M; Na, 0.1 to 4 M; SO_4, 0 to 1.0 M; acidity, 0 to 6 M; wastes from uranium-aluminum fuels contain about 0.65 M

Table 4.6. Estimated Volumes of Low- and Intermediate-Level Wastes[a]

Calendar Year Ending	Gallons per Year	Accumulated Gallons
	Intermediate-Level Wastes[b]	
1970	31,000	31,000
1980	777,000	3.5×10^6
1990	2.6×10^6	2.0×10^7
2000	3.2×10^6	4.9×10^7
	Low-Level Wastes[c]	
1970	1.6×10^6	
1980	3.9×10^7	
1990	1.5×10^8	
2000	1.4×10^8	

[a]Based on fuel processing projections of Phase 3, Case 42, Systems Analysis Task Force (April 1968).

[b]Based on generation of 200 gal of intermediate-level waste per metric ton of fuel processed.

[c]Based on generation of 10,000 gal of low-level waste per metric ton of fuel processed.

Table 4.7. Solid Wastes from Spent Fuel Processing[a,e]

	Calendar Year			
	1970	1980	2000	2020
Volume of cladding waste[b]				
Annual, 10^3 ft^3	0.3	8.3	87	200
Accumulated, 10^3 ft^3	0.3	37	1,030	3,800
Total volume of solid waste[c]				
Annual, 10^6 ft^3	0.03	0.8	3	7
Accumulated, 10^6 ft^3	0.03	4	49	140
Burial ground area[d,e]				
Annual, acres	0.6	16	64	140
Accumulated, acres	0.6	70	970	2,800

[a]Based on fuel processing projections of Phase 3, Case 42, Systems Analysis Task Force (April 1968).

[b]Based on 2.1 ft^3 of cladding hulls per ton of LWR fuel processed, and 8.7 ft^3 of cladding hardware per ton of LMFBR mixed core and blankets processed.

[c]Based on an average volume of 200 ft^3 of solid wastes per ton of fuel processed.

[d]Based on burial of 50,000 ft^3 solid waste per acre of burial ground.

[e]Wastes containing significant amounts of long-lived alpha nuclides would be treated as high-level wastes.

$Al(NO_3)_2$. LMFBR wastes are expected to be chemically similar to LWR wastes, and will contain Fe, Ni, and Cr, but no sulfate, from partial dissolution of the stainless steel fuel cladding.

Tank Storage of High-Level Aqueous Wastes. - High-level wastes, which originate mainly from the first cycle of solvent extraction, contain more than 99.9% of the nonvolatile fission products. The present practice is to concentrate and store these wastes on an interim basis in underground tanks that are equipped with devices for removing the decay heat. More than 80 million gallons of such wastes are now in storage at various AEC production sites; and, in spite of close surveillance, there have been 14 known instances of tank failure. Although the resultant waste releases have caused no radiation exposures to personnel and have been retained in the near vicinity of the tanks, it is clear that many of the liquid-waste storage facilities now in existence do not possess the necessary degree of integrity to permit permanent or long-term storage. Storage of liquid wastes produced in future power-reactor fuel processing will be even more difficult than storage of current production wastes because of the higher heat-generation rates and the significant rates of radiolytic hydrogen production of the former. It should, nevertheless, be possible to store them safely for a limited time and at an acceptable cost, provided adequate engineered safeguards are built into the storage systems.

Solidification of High-Level Aqueous Wastes. - The alternative to long-term or "perpetual" storage of wastes in tanks is conversion of the wastes to thermally and radiolytically stable solids of low solubility for burial in selected geologic formations or storage in man-made vaults. Processes for converting these wastes to solids are being developed both in the United States and abroad. The four principal solidification methods used in the United States are the pot, spray, phosphate glass, and fluidized-bed processes, which were developed at Oak Ridge National Laboratory (ORNL),[5] Pacific Northwest Laboratory (PNL),[6] Brookhaven National Laboratory (BNL),[7] and the Idaho Chemical Processing Plant (ICPP),[8] respectively. The pot, spray, and phosphate glass processes have been undergoing demonstration for the AEC at full radioactivity level, on an industrial scale, at PNL since November 1966.[9] Demonstration is scheduled to be complete in 1970. Since 1963, the fluidized-bed

process has been undergoing demonstration at the ICPP in a large-capacity plant operating with intermediate-level radioactive wastes. Within the next few years, the AEC's waste solidification development program for currently known concepts will be completed. Each process will have been demonstrated by using wastes from advanced high-exposure fuels, and the effects of severe temperature and radiation on the properties of the solidified waste products will have been measured and evaluated. These data will provide reliable bases for the design and safe operation of waste solidification plants.

In the pot calcination process, the concentrated aqueous wastes are evaporated to dryness and then calcined in a stainless steel pot. When filled with solids, the pot is sealed by welding and becomes the final disposal container. In the spray calcination process, the waste is first mixed with glass-forming additives. The resulting mixture is spray-calcined, liquefied in a platinum melter, and the glass product is collected in a receiver. In the phosphate glass process, the glass-forming additives are added to the waste, the nitric acid is removed by distillation, the remaining volatiles are removed in a platinum melter, and the glass product is collected in a receiver. In the fluidized-bed calcination process, the wastes are calcined in a fluidized bed and the solid granular product is subsequently discharged to a storage bin or a pot. After interim storage, the pots from each solidification process could be transported to a final disposal site. The processing rates for the three processes, as demonstrated in the industrial-scale tests with high-level radioactive wastes at PNL, were equivalent to 0.75 and 0.9 metric ton of uranium per day for 8-in.- and 12-in.-diam pots for the pot process, 0.65 metric ton/day for the spray process, and 0.67 ton/day for the phosphate glass process. The fluidized-bed process has been used to treat aluminum-type waste that is less radioactive, by two or three orders of magnitude, than high-level Purex wastes. If a similar solid production rate is assumed for Purex waste, the equivalent processing rate would be about 9.0 tons of uranium per day.

The characteristics of the solidified wastes for the four processes are summarized in Table 4.8.[4] The thermal conductivity and the heat

Table 4.8. Characteristics of Solidified High-Level Wastes[a]

	Solidification Method			
	Pot Calcine	Spray Melt	Phosphate Glass	Fluidized Bed
Form	Monolithic	Monolithic	Monolithic	Granular
Description	Scale	Microcrystalline[b]	Glass	Amorphous[c]
Chemical composition, mole %				
Fission product oxides	15-~80	5-30	5-25	5-50[a]
Inert metal oxides	10-50	40-50	10-30	10->90
Sulfur oxides (if in waste)	0-40	0-40	0	0-40
Phosphorus oxides	~0	25-40	~60	~0
Bulk density, g/ml	1.1-1.5	2.7-3.2	2.7-3.0	1.0-1.7
Thermal Conductivity, Btu hr^{-1} ft^{-1} $°F^{-1}$	0.15-0.25	0.4-0.8	0.4-1.0	0.10-0.25
Maximum heat generation rate, w per liter of solid[d,e]	100 (85)[f]	200 (35)	210 (315)[g]	70 (0.5)
Solubility in cold water, g cm^{-2} day^{-1}	$1.0-10^{-1}$	$10^{-3}-10^{-6}$	$10^{-4}-10^{-7}$	$1.0-10^{-1}$
Hardness	Soft	Hard	Very hard	Hard

Table 4.8. (Cont'd.)

	Solidification Method			
	Pot Calcine	Spray Melt	Phosphate Glass	Fluidized Bed
Friability	Crumbly	Tough	Brittle	Moderate
Residual nitrate, wt % of product	≤0.05	≤0.005	≤0.005	≤4.0

[a]Composition ranges for fluidized bed are also for Purex waste and are estimated.

[b]Glassy products can also be made with some difficulty.

[c]Microcrystalline products can also be made.

[d]Calculated values for storage in air in 8-in.-diam cylindrical pots to maintain pot center-line temperatures at less than 900°C and pot wall temperatures at less than 425°C. Average k values were used.

[e]Values in parentheses are the highest values achieved in pilot-plant operations as of May 1968.

[f]8-in.-diam pot

[g]6-in.-diam pot.

generation rates are two of the most important limiting parameters. The maximum permissible heat generation rates in 8-in.-diam disposal pots are approximately 100 w/liter for calcines and approximatly 220 w/liter for glass products in order to maintain the center-line temperature at 900°C and the wall temperature at 425°C. These values are equivalent to those obtained for wastes from LWR fuels that had been irradiated to 20,000 Mwd/metric ton and cooled 360 and 180 days, respectively. Even longer cooling times would be required before wastes from higher-burnup fuels could be solidified in 8-in.-diam pots. For the case of phosphate glass in 6-in.-diam pots (probably the minimum economical size), the maximum heat generation rate is 402 w/liter in order for the center-line and the surface temperatures to be maintained at 938°C and 425°C, respectively. This value corresponds to an LWR fuel irradiated to 40,000 Mwd/metric ton and cooled for approximately 66 days. The calculation[10] assumes thermal conductivities for phosphate glass of 0.58 Btu hr^{-1} ft^{-1} $°F^{-1}$ at 200°C, and 0.84 Btu hr^{-1} ft^{-1} $°F^{-1}$ at 1000°C.[11] The waste from Atomics International Reference Oxide LMFBR fuel, irradiated to an average burnup (core plus blankets) of 33,000 Mwd/metric ton and cooled for 30 days, would have a heat generation rate of 860 w/liter when the final volume is 1 ft^3 per 10,000 Mwd per metric ton. A total cooling time of 90 days would be required in this case before the phosphate glass waste could be encapsulated in 6-in.-diam pots, to ensure that the stipulated thermal conditions were not exceeded. Similarly, pot-calcined LMFBR waste with a thermal conductivity of 0.26 Btu hr^{-1} ft^{-1} $°F^{-1}$ could be solidified in 6-in.-diam pots, after 2.7 or 1.4 years of cooling, to give a final volume of 0.81 or 1.84 ft^3/metric ton, respectively.[3] These values illustrate the importance of thermal conductivity in determining the size of the storage vessel.

Thus, it is apparent that the current waste solidification methods, which are in an advanced stage of development, can be used to solidify LMFBR wastes if the wastes remain in liquid storage long enough to reduce the heat production rates to the required levels.

Dilution of the solid product with inert solids is another method for decreasing the heat generation rate. For example, in its proposed new processing plant, the General Electric Company plans to dilute the liquid waste with aluminum nitrate prior to calcining. This treatment will limit the heat generation rate in the final product in 6-in.-diam pots to 200 w/liter.

A systematic survey is being made of the wastes expected from aqueous processing of fuels from light-water, fast breeder, and gas-cooled reactors to determine the most economical approaches to their management within the technology that is being developed.[3] While it is recognized that immediate or early solidification (immobilization) is desirable on the basis of safety, a number of variables must be correlated to optimize both safety and economics. These variables include thermal conductivity, cooling time in liquid storage, number and size of pots, dilution with solids, the combination of various intermediate and high-level wastes, and cost of liquid storage vs storage as solids. Development work to characterize these wastes and provide the data required for the survey is under way.

A method of increasing the thermal conductivity of the product by dispersing the waste solids in glass matrix is also being studied at ORNL.[3,12] Simulated solid waste powder from processing high-temperature gas-cooled reactor fuel was dispersed in a lead silicate glass at 800°C. The measured thermal conductivity of this dispersion was 1.73 Btu hr^{-1} ft^{-1} $°F^{-1}$ at 600°C, a fourfold increase over that of the waste powder. Simulated LWR and LMFBR wastes have also been dispersed in a low-melting (800-900°C), noncorrosive, borosilicate glass matrix to produce products containing up to 60 wt % waste solids. Recent results indicate that the pot calcination system can be used to prepare dispersion products and thus eliminate the high temperatures and the platinum melter customarily used to prepare glassy products. If the thermal conductivity proves to be as high as 0.94 or 1.4 Btu hr^{-1} ft^{-1} $°F^{-1}$, the minumum cooling times for solidification in 6-in.-diam pots will be 0.16 or 0.53 year for final volumes of 4.89 or 1.84 ft^3 per 33,000 Mwd per metric ton.

Immobilization of Gaseous High-Level Wastes. - Large inventories of radioactive gases, such as Kr, I_2, ^3H, and Xe, will be collected in future fuel processing plants. Some methods for immobilizing these gases is required prior to their transport to a government area for permanent disposal. The current practice in commercial plants involves eliminating most of the short-lived nuclides by decay prior to processing and either retaining the remaining short- and long-lived nuclides in units such as caustic scrubbers, silver nitrate towers, charcoal beds, or high-pressure cylinders, or discharging them to the environment. Present methods that can be considered for disposal of noble fission product gases include: (1) controlled discharge to the atmosphere, (2) containment and storage in pressurized gas cylinders,[13] (3) underground injection in carefully selected geological formations,[14] and (4) new proposals for encapsulation.[15]

Release to the atmosphere within safety limits becomes more difficult as nuclear installations become larger and more numerous. Disadvantages of injection into the ground include: (1) uncertainty regarding eventual contamination of the environment, and (2) the probable requirement for disposal of all wastes in a government area. The use of cylinder storage allows close surveillance but involves the possibility of a sudden dangerous discharge in the event of a rupture or a leak in the container. It is believed that modification of cylinder storage, by adding at least one extra envelope of containment, represents the safest presently available method for disposal of gases.[15] This method retains the basic advantages of compression to a small volume and storage under carefully controlled conditions.

The amounts of volatile or potentially volatile fission products per metric ton of reference fuel (Table 4.9) and other factors must be considered in selecting a disposal method. The total weight of all iodine fission product isotopes increases from 429 to 440 g per metric ton of fuel if the precursors are not separated. Greater than 99.7% of the iodine is stable ^{127}I and long-lived ^{129}I at 30 days cooling. The radioactive xenon has decayed to a negligible level after 150 days, and its volume remains at about 13 times that of krypton. Therefore, it may be desirable to separate and discharge the xenon and hold only the krypton for long-term storage.

Table 4.9. Amounts of Volatile or Potentially Volatile Fission Products per Metric Ton of Homogenized LMFBR Fuel[a]

Isotope	30 Days			150 Days			30 Years		
	Ci	g	liters(STP)[b]	Ci	g	liters(STP)[b]	Ci	g	liters(STP)[b]
^{3}H (12.26 years)	930	9.60×10^{-2}	0.78	913	9.42×10^{-2}	0.77	172	1.78×10^{-3}	0.015
^{83}Kr (stable)	0	67.1	19.8	0	67.1	19.8	0	67.1	19.8
^{85}Kr (10.76 years)	1.03×10^{4}	26.3	7.6	1.01×10^{4}	25.7	7.4	1.49×10^{3}	3.82	1.1
Total[c] Kr	1.03×10^{4}	314.	91.9	1.01×10^{4}	314.	91.9	1.49×10^{3}	292.	85.5
^{127}I (stable)	0	111.	10.7	0	114.	11.0	0	117.	11.3
^{129}I (1.6×10^{7} years)	5.16×10^{-2}	317.	30.0	5.26×10^{-2}	322.	30.5	5.27×10^{-2}	323.	30.6
^{131}I (8.05 days)	1.41×10^{4}	1.14	0.1	4.58	3.7×10^{-5}	0.0	0	0	0.0
Total[c] I	1.45×10^{4}	429.	40.8	4.63	437.	41.5	5.27×10^{-2}	440.	41.9
^{131m}Xe (12 days)	6.28×10^{3}	7.43×10^{-2}	0.014	7.47	8.84×10^{-5}	0	0	0	0
^{133}Xe (5.27 days)	6.52×10^{4}	3.50×10^{-1}	0.064	9.11×10^{-3}	4.89×10^{-8}	0	0	0	0
Total[c] Xe	7.14×10^{4}	4280.	793.	7.48	4280.	793.	0	4280.	793.

[a]Atomics International Reference Fuel (see Sect. 3.1) combined core and blankets; burnup, 33,000 Mwd (thermal)/metric ton.

[b]Gas volume calculated for ^{3}H as $H^{3}H$ and I as I_2. Here, STP means 298°K and 760 mm Hg pressure.

[c]Total for all isotopes.

The combined krypton and xenon from processing five metric tons of fuel per day would fill about 76% of a standard pressurized "K" cylinder (ICC-3A 2000).

Methods proposed[3,15] for the disposal of gaseous wastes include conversion of iodine and tritium to stable solid or liquid compounds such as NaI, ^3HHO, or Na^3HSO_4 for storage, and storage of krypton as a gas. The solid waste products would be sealed in a stainless steel primary containment vessel (such as that used for high-level wastes) during temporary storage and shipment to a government repository. Both primary and secondary containment, as opposed to simple pressurized tank storage, seem desirable for the more mobile gases. Several alternative primary containment methods, each of which uses a stainless steel pot or pressure cylinder as a secondary containment vessel, were considered. These include entrapment in molecular sieves or clathrates, dispersion as a foam in glass, plastic, or metal media, and dispersion as microampuls in similar media. Some of these concepts have been tested in preliminary experimental work.[3,15]

Treatment for Low- and Intermediate-Level Wastes. - New waste management concepts indicate that intermediate-level wastes will probably be considered only as wastes which are transitory in an installation, and that they will be converted to, or combined with, high- or low-level wastes. Many intermediate-level wastes will contain significant amounts of long-lived alpha nuclides and, as a result, would be classified as high-level wastes. For example, a loss of 0.01% of the plutonium in a 5-metric ton-per-day plant (20% PuO_2--80% UO_2) to the intermediate-level waste would amount to 36 kg of plutonium per year. The chemical and engineering effects of combining these wastes with the normal high-level wastes are now being studied at ORNL. Initial estimates indicate that this procedure should not increase the cost of waste management significantly.[3]

Low- and intermediate-level wastes that contain negligible amounts of long-lived alpha nuclides can be handled by techniques similar to those presently in use. Liquid wastes are generally treated by distillation, flocculation, or ion exchange, and the residues are then stored or buried

along with miscellaneous solid wastes. New techniques are also available. The disposal of intermediate-level waste-cement slurries in deep geologic formations by hydrofracturing has been developed and demonstrated at ORNL.[16] This method requires a favorable geology. However, it could not be used at private installations if regulations require disposal only on government-owned land.

Another technique developed at ORNL is the dispersion of intermediate-level waste residues in asphalt or polyethylene in order to decrease the solubility of the final product and to decrease the potential for accidental release of radioactivity to the environment during subsequent storage or burial. Dispersion in asphalt or polyethylene decreases the leach rate of radionuclides from solid residues by factors of 10^4 to 10^6.[3,17,18] The final residues could be collected in inexpensive containers and shipped to a government-controlled disposal area. Further development of this method has been deferred at ORNL. It was concluded that asphalt and polyethylene have satisfactory stability to radiation (i.e., about 10^8 and $>10^9$ rads, respectively) and that asphalt and polyethylene are good media for incorporating organic and inorganic intermediate-level wastes that do not contain oxidizing materials such as nitrates. Asphalt is not recommended for wastes which contain oxidants. Polyethylene is probably acceptable for use with oxidizing salts, but further studies are required to assess the safety of this system. Polyethylene is equal to, or better than, asphalt on most bases. Although polyethylene is more expensive than asphalt, the cost of the dispersion medium is probably not a significant factor in the cost of operating a nuclear installation.

Shipment of Solidified Wastes.[19] - A processor can exercise considerably more control over the solid wastes leaving his plant than over the spent fuel entering it. For example, cooling times for wastes can easily be varied without incurring the economic penalties to which spent fuel is subject. In addition, waste containers can be designed for shipment via either truck or rail, whereas there may be little or no choice as to the means for transporting spent fuel. The waste product will be doubly contained, first by a welded stainless steel container and then by the shipping cask itself. The calcined or glass waste product is relatively immobile;

although the 30-ft impact accident condition would create some fracturing in the product, it would be of little consequence. The 1475°F fire accident condition could cause the center-line temperature of calcined wastes to increase above 900°C; however, the consequences of this thermal transient do not appear severe. Pressure increases would be small, certainly within the resistance capabilities of the steel pot whose maximum temperature will be in the range of 500 to 600°C. In short, the degree of control available over solid waste shipments, coupled with the fact that the fission products are in a relatively nondispersible form, suggests that such shipments should be quite safe.

Disposal in Natural Formations.[20] - The most promising method for disposing of solidified high-level wastes is believed to involve final storage in natural salt formations. Therefore, a 19-month demonstration disposal of high-level radioactive waste solids was carried out in a salt mine at Lyons, Kansas. Some reactor fuel was used in lieu of actual solidified wastes. In the course of this program, most of the technical problems related to disposal in salt were resolved. The feasibility and the safety of handling highly radioactive materials in an underground environment were demonstrated; the stability of salt under the effects of heat and radiation was shown; and data on the creep and plastic flow characteristics of salt were obtained. Cost studies indicate that this method is economically acceptable. The 2000 acres of salt area that may be committed for disposal purposes by the year 2000 is only a very small fraction of the 0.5 million square miles that are underlain by salt in the United States.

Dry openings that could be utilized for the storage of radioactive solid wastes can be excavated in rocks other than salt; however, extensive investigations would be needed to define more precisely the geological conditions that would determine the usefulness of local sites within the most desirable geographic regions and the effects of heat and radiation on the enclosed rock media.

4.9.4 Waste Management Costs

Economic analyses made of interim liquid storage, pot calcination,

interim solid storage, shipment, and disposal in salt mines indicate that these operations can be accomplished for about 0.02 mill per kilowatt-hour of electricity produced from the original fuel (Table 4.10).[21] For comparison, "perpetual" tank storage of the wastes in liquid form was estimated to cost from 0.017 to 0.018 mill/kwhr.[22] While the costs of both methods are acceptable in that they represent only about 1% of the total fuel-cycle cost anticipated for 4-mill/kwhr nuclear power (10% of current processing costs), "perpetual" storage in tanks does not offer safety commensurate with that of storage of solidified wastes in salt mines.

These costs are based on a plant that would process the fuel from an installed nuclear capacity of 22,400 electrical megawatts (1.56×10^{11} kwhr/year generated), and were computed on the basis that all necessary capital and operating expenses are recovered during a 20-year period of plant operation. Waste management unit costs decrease only slowly with increasing plant size, perhaps 30% as the size increases by a factor of 10. Thus, in 1970, waste management may contribute 10% of a total processing cost of 0.20 mill/kwhr(electrical). However, in the year 2010, it is estimated that waste management would contribute about 25% of a predicted fuel processing cost of 0.05 mill/kwhr(electrical).

The cost of interim storage of solidified waste is one-third to one-fifth that for storage of the same wastes as liquids, and the cost, per unit volume, of 6-in.-diam containers is four times that of 2-ft-diam containers. Thus, there is a cost incentive, as well as a safety incentive, for producing solids with higher thermal conductivities at the earliest practical time.

The above cost estimates for waste management were made for a government-financed project, using 1962 costs and 4% interest on money. They are being revised to reflect private financing using 1969 costs. The final costs are expected to increase from 0.02 to 0.03-0.05 mill/kwhr (electrical).[2] The processing and waste management cost estimates should probably also be revised upward by about 10% to allow for improved containment systems costs, including provisions for enhanced removal of rare gases and iodine, improved containment of internal explosions, and earthquake-resistant design and construction. The use of improved containment systems appears to be more economical than accepting the extremely large and

Table 4.10. Management Costs for Acid Purex and Thorex Wastes[a]

Management Operation	Interim Solid Storage Time			
	2 Years		30 Years	
	10^{-3} mill per kwhr(e)	$/ton[b]	10^{-3} mill per kwhr(e)	$/ton[b]
Interim liquid storage	0	0	0	0
Pot calcination in 6-in.-diam pots at 1/3 year	10.7	1640	10.7	1640
Interim solid storage	1.7	260	4.4	760
Shipment (2000 miles round trip)	1.8	280	0.4	60
Disposal in salt	8.7	1340	2.0	310
Total	22.9	3520	17.5	2770

[a]Present-valued at 4%.

[b]Based on a fuel exposure of 20,000 Mwd/metric ton.

remote sites that would otherwise be required for large processing plants, especially for short-cooled fuel.

4.10 Off-Gas Treatment - Task 10

4.10.1 Introduction

The routine discharge of radioactive gases and effluents is one of the major problems facing the nuclear fuel processing industry today. This problem will become more acute as we progress to higher burnups, larger plants, and, perhaps, shorter-cooled fuels.

The radionuclides of most importance in the routine atmospheric discharge from spent-fuel processing plants are ^{85}Kr, ^{133}Xe, ^{3}H, ^{131}I, and ^{129}I. Results of our studies indicate that worldwide pollution hazards from the release of these radionuclides are avoided and that plant personnel are protected by the necessary expedient of ensuring appropriately low exposures of the public near the site boundary.

Table 4.11 shows concentrations of these radionuclides in typical light-water reactor fuel, cooled 5 months, and in mixed core-blanket fuel of the Reference Oxide LMFBR, cooled 1, 2, and 5 months. Concentrations of these nuclides are determined, primarily, by the burnup of the fuel, the specific power to which the fuel was exposed, and the postirradiation decay time, since the neutron absorption cross sections are low and the yields from fast and thermal fissions of uranium and plutonium are not greatly different. The concentrations of the long-lived nuclides $^{85}Kr(t_{1/2} = 10.8$ years), $^{3}H(t_{1/2} = 12.3$ years), and $^{129}I(t_{1/2} = 1.6 \times 10^{7}$ years) are about 0.3, 0.025, and 1.5×10^{-6} curie per megawatt-day of burnup, respectively. After 30 days of cooling, the concentrations of the short-lived nuclides $^{133}Xe(t_{1/2} = 5.3$ days) and $^{131}I(t_{1/2} = 8$ days) are about 1200 and 2200 curies per megawatt of reactor power, respectively.

The maximum acceptable average annual concentrations, in air, of the above gaseous radionuclides at the site boundary of private plants are one-third of the maximum permissible concentrations (MPC_A) as stipulated by the Code of Federal Regulations, Title 10, Part 20, Appendix B, Table II.

Table 4.11. Concentrations of Volatile Fission Products in Spent Reactor Fuel

	Reactor			
	LWR	LMFBR (Core + Blanket)		
Fuel				
Burnup (Mwd/metric ton)	33,000		33,000	
Specific power (Mw/metric ton)	30		58	
Decay (days)	150	30	60	150
Concentration (Ci/metric ton)				
^{85}Kr	11,200	10,300	10,200	10,000
^{133}Xe	0.0053	74,400	1430	0.010
^{3}H	688	932	928	915
^{131}I	2.16	139,000	10,500	4.52
^{129}I	0.038	0.0528	0.0533	0.0537

These MPC_A values (reduced by the factor of 3) in Ci/m^3 are: ^{85}Kr or ^{133}Xe, 1×10^{-7}; ^{3}H, 6.7×10^{-8}; ^{131}I, 3.3×10^{-11}; and ^{129}I, 6.7×10^{-12}. There is little concern regarding reconcentration of noble gases, and probably tritium, in the environment and pathways back to man. For example, a concentration of 0.67×10^{-8} Ci of tritium per cubic meter in air at 70°F and 40% relative humidity corresponds to a concentration of 9×10^{-3} Ci/m^3 in the water vapor, only a factor of 3 higher than the maximum limit imposed by the above regulations for continuous ingestion of water by the public. It is unlikely, therefore, that mechanisms such as fog formation or rain would cause hazardous reconcentration.

Additional factors must be considered for ^{131}I and ^{129}I. Experimental evidence[1] has indicated that the ^{131}I limit should be reduced by an additional factor of 700, that is, to $(1 \times 10^{-10})(3)^{-1}(700)^{-1} = 4.8 \times 10^{-14}$ Ci/m^3, as a means of accounting for reconcentration in the grass-cow-milk pathway to the thyroids of small children. A similar limit does not presently exist for ^{129}I because of uncertainties in the manner in which it will accumulate in grass over long periods. We have tentatively assumed a value of $(2 \times 10^{-11})(3)^{-1}(700)^{-1}(10)^{-1} = 1.0 \times 10^{-15} Ci/m^3$ as the maximum acceptable concentration for ^{129}I. The extra factor of 10 was derived by using the radiochemical half-lives of ^{131}I(8 days) and ^{129}I(1.6×10^7 years), and an assumed half-life of 80 days for the residence of iodine on the ground (i.e., the ratio of half-lives, 80/8 = 10). The residence half-life is based on assumptions for loss from resuspension, percolation, erosion, cropping, etc. The weight of iodine on the surface of the ground at equilibrium, 0.5×10^{-7} g/m^2, also appears to be adequately low. On this basis, the acceptable concentration of ^{129}I in air at the site boundary is about 50 times lower than that for ^{131}I, but the ^{131}I will be controlling for typical reactor fuels that have cooled for about 5 months or less since the number of curies of ^{131}I is much larger than that of ^{129}I.

Table 4.12 shows the approximate decontamination factors (DF's) that are required for these volatile effluents in order to achieve the maximum acceptable concentrations at a site boundary that is 2 to 3 km distant, in

Table 4.12. Approximate Decontamination Factors Required for Volatile Effluents from Fuel Processing Plants

Site boundary distances of 2 to 3 km from 100-m stack.

Type of Effluent	Decay Time (days)	Required DF's for Plant Capacity (metric tons/day) of:			
		1	5	10	40
Noble gas	30[a]	2	10	20	90
	$\geqq$ 60[a,b]	-	1.3	2.6	10
Tritium	$\geqq$ 1[a]	-	-	-	1.4
Iodine	30[b]	7×10^6	4×10^7	7×10^7	3×10^8
	60[b]	6×10^5	3×10^6	6×10^6	2×10^7
	150[b]	2×10^2	1×10^3	2×10^3	9×10^3
	150[c]	1×10^2	5×10^2	1×10^3	4×10^3

[a]Burnup = 33,000 Mwd/metric ton.

[b]Specific power = 58 Mw/metric ton.

[c]Specific power = 30 Mw/metric ton.

the direction of the prevailing winds, from a 100-m stack. These conditions of downwind distance and stack height were assumed to result in an average annual downwind concentration per unit of emission rate of 3×10^{-7} $(Ci/m^3)/(Ci/sec)$ at the site boundary. A comparison of this atmospheric concentration parameter as a function of maximum downwind distance, as estimated for the ORNL, Hanford,[2] NRTS,[3] and Savannah River sites, is presented in Fig. 4.14. The ORNL, Hanford, and NRTS data in this figure are based on meteorological calculations averaged over annual weather conditions, but are known to be reasonable, based on long-term environmental monitoring studies. The Savannah River data are derived from air sampling studies for ^{131}I at the site boundary over a period of a year. The Savannah River data reflect the effect of depletion of the plume[4] that may result from the deposition of radionuclides, particularly for an active species such as I_2.

The 2- to 3-km distance from the site boundary was chosen as a reasonable upper limit for private processing sites. The average annual downwind concentration generally attains a maximum somewhat short of this distance and then decreases, roughly, in direct proportion to the distance. Generally, it will be more economical to decontaminate the gas stream by the required factor rather than to increase the distance to the site boundary. Our studies indicate that site radii of 2 to 3 km are adequate to render the effects of conceivable accidental releases acceptable.

The annual allowable discharge rates from the 100-m stack that would be required in order to achieve the maximum acceptable concentrations (individually) at the 2- to 3-km-distant site boundary are about 10,000,000 Ci of the noble gases, 7,000,000 Ci of ^3H, or 5 Ci of iodine. If the fuel is to be processed after only 30 days of cooling, the removal of noble gas should begin for a plant capacity of about 0.4 ton/day; however, if the cooling period is at least 60 days, removal should begin for a plant capacity of about 4 tons/day. Actually, the required DF's for noble gases are not very large, even in high-capacity plants. On the same basis, removal of ^3H is not required for plants having capacities of less than 20 to 30 tons/day, provided the ^3H is volatilized up the stack. The required iodine DF's are in the range of 10^7 to 10^8 for 30-day-cooled

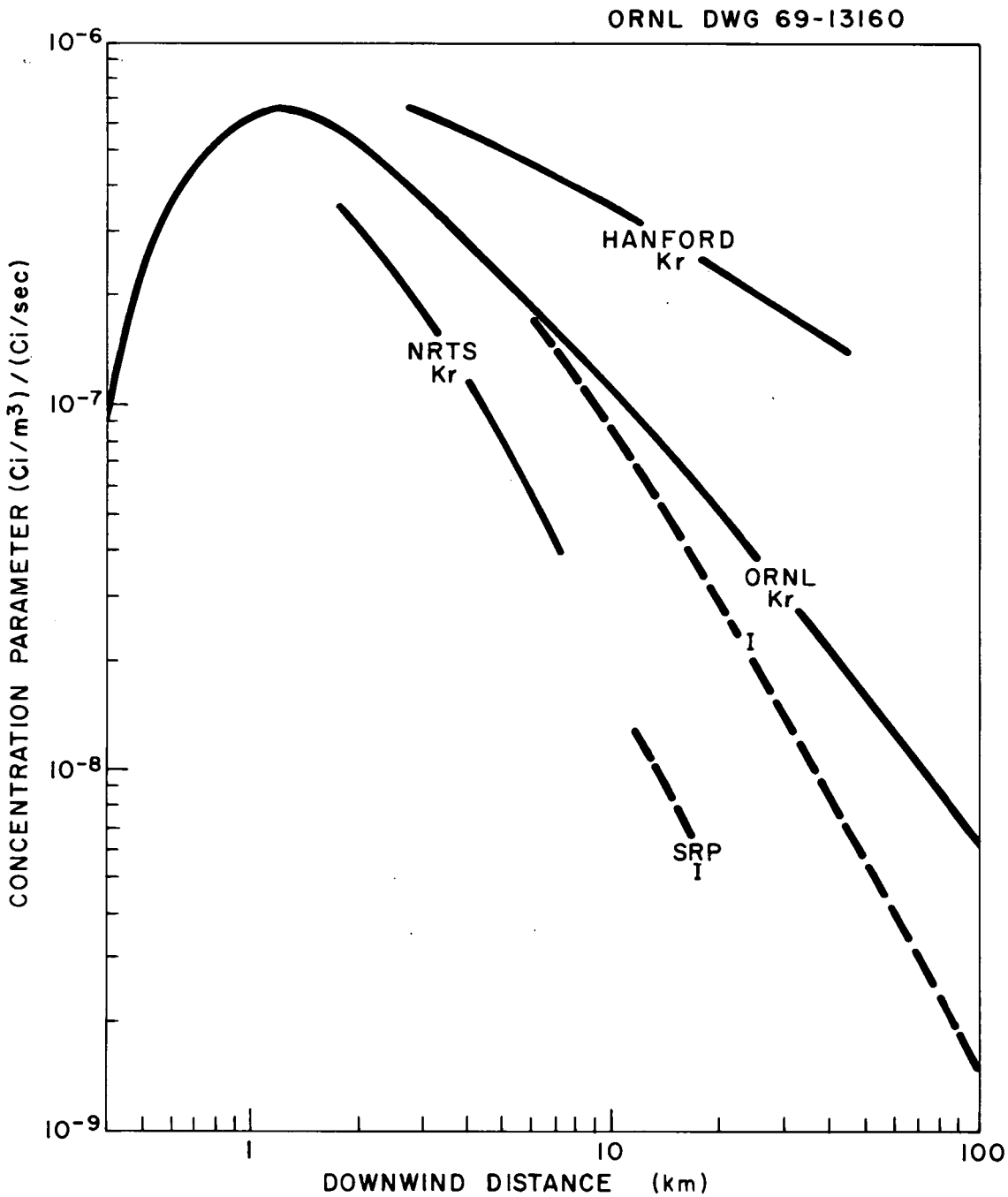

Fig. 4.14. Effect of Downwind Distance on the Average Annual Downwind Ground Concentration per Unit Emission Rate from a 100-m-tall Stack.

fuel, 10^6 to 10^7 for 60-day-cooled fuel, and, in the range of current technology, 100 to 1000 for 150-day-cooled fuel.

The limits for the release of radionuclides quoted in the preceding discussion are based on the Code of Federal Regulations. However, the licensing policy of the AEC has tended to be more conservative than indicated by these limits, and the trend in Federal agencies has been toward more stringent controls such that reductions in release limits may occur. Therefore, gas treatment methods should be developed which, within reasonable economic limits, can meet stricter requirements if necessary.

4.10.2 Iodine Retention

The development of iodine removal systems for fuel processing plants has only proceeded as far as that required for early LWR fuels. Present processing plants do not provide the degree of decontamination or the positive control that would be required for processing LMFBR fuels cooled less than 150 days. Comparisons between units in different operating plants are difficult and almost meaningless because of the diverse conditions encountered in, for example, process and cell off-gas routing, operating procedures, and off-gas composition (particulates, organics, etc.).

At present, there is not sufficient knowledge of the chemical behavior of iodine under the various conditions encountered in a fuel processing plant to design an off-gas system required for short-cooled LMFBR fuels. We know that iodine exists in at least three oxidation states and forms a variety of compounds with various impurities, some of which are volatile. It will be necessary to systematically study the chemistry of iodine in the shearing step, the voloxidizer, the dissolution step, the acid and caustic scrubber, the charcoal traps, and the silver bed, using a range of temperatures, off-gas compositions, and oxidation-reduction conditions. Only after such studies have been made can processes be chosen to remove and positively retain iodine to the degree required for short-cooled LMFBR fuel processing.

In addition, it will be necessary to make engineering-scale tests of equipment and process design, and finally hot-cell and hot pilot-plant-scale tests, to prove the reliability of the iodine retention system.

Present Processing Systems - Hanford and Savannah River. - The treatment units are similar at both installations; in each case, the overall removal efficiency does not greatly exceed that required for off-gases from relatively low-burnup fuel cooled 120 to 150 days. At the Hanford Purex Plant, the equipment sequence after the dissolver is as follows: up- or downdraft condenser, silver reactor, filter, two bubble-cap nitric acid scrubbers, filter, and stack. About 80% of the iodine is volatilized from the dissolver. Approximately 80% of the iodine released from the stack originates in the vessel off-gas system, which contains organic vapors. Only about 20% comes from the separate dissolver off-gas system. In the vessel off-gas system, as opposed to the dissolver off-gas system, the silver reactor is not operated at a sufficiently high temperature to obtain efficient iodine removal because of the possibility of an explosion with the organic vapors.[5] The organic vapors also form methyl and other alkyl iodides, which are not efficiently removed.[5] The silver reactors[6] produce iodine DF's of 100 to 1000, and the two nitric acid scrubbers[5] provide a total DF of about 150. The iodine concentration downstream of the silver reactors varies from 25 to 80 $\mu Ci/ft^3$. Previously, a caustic scrubber was used instead of the second acid scrubber; both units showed about the same efficiency, namely, a DF of about 3.

Dounreay. - Normally, the facility at Dounreay uses an updraft packed condenser followed by a water scrubber, which gives an iodine DF of 2.[7,8] A 1 M HNO_3--0.001 M $Hg(NO_3)_2$ scrubber was installed for use with 90-day-cooled 9 wt % Mo--91% U fast reactor fuels. About 95% of the iodine was volatilized from the dissolver, and the iodine DF across the scrubber was 200 to 250. The low decontamination factor was attributed to the presence of organic iodides.

Windscale and Eurochemic. - These plants were designed to use fumeless (O_2 addition) dissolvers, caustic scrubbers, and charcoal absorbers, but they have not had a sufficient concentration of iodine present to obtain accurate measurements on the efficiency of iodine removal.

ICPP. - At the Idaho Chemical Processing Plant, short-cooled fuels have been processed (RaLa process) to produce ^{137}Ba, with up to 2.5×10^4 Ci of iodine present in a dissolving unit. The equipment sequence is: condenser, $Hg(NO_3)_2$--HNO_3 scrubber, charcoal bed, and micrometallic filters (20 μ).[9-12] The overall iodine DF was 3×10^4 as measured at the stack. Since the sampling stations and cells vent directly to the stack, the decontamination efficiency of the units may be higher. The efficiency of the mercury scrubber was not measured, but the charcoal bed and filter showed an overall DF of 38. The charcoal bed removed 99.99% of the gaseous iodine but only 88% of the particulate iodine. A caustic scrubber (1 M NaOH), used initially, removed 90% of the iodine; when the caustic contained 0.1 M thiosulfate, about 97% was removed.

Nuclear Fuel Services.[13] - The initial equipment sequence was as follows: downdraft condenser, acid scrubber, silver reactor, absolute filter, and stack. An iodine DF of about 13 is required with present 1.5-year-cooled fuels, but a DF of 500 to 1000 will be required with 150-day-cooled, high-burnup fuels. The silver reactor has been replaced with a caustic scrubber largely because of the cost of silver but also because of operating problems. Significant weights of iodine (^{129}I) are present in high-burnup fuels, and silver requirements rise accordingly. An extra filter to assist in meeting particulate discharge requirements is also being considered, along with a silver zeolite bed for removal of small amounts of organic iodides.

TRU. - The Transuranium Processing Plant (TRU) at ORNL processes plutonium, curium, and californium targets from the HFIR after 1 to 10 days of cooling. Each target contains about 200 Ci of ^{131}I at discharge. The steps used to process targets are: (1) dissolution of the aluminum (jacket and matrix material), using a NaOH--$NaNO_3$ solution; (2) dissolution of the actinide oxide, using 6 M HCl; (3) separation and recovery of the unburned plutonium via batch extraction from 6 M HCl, using 1 M HDEHP in DEB; (4) decontamination of the transplutonium elements, using the Tramex solvent extraction process; (5) separation of americium and curium from the transcurium elements, using LiCl-based anion exchange; (6) further purification of plutonium, using a second cycle of the Pubex

process; and (7) separation and purification of the transcurium elements, using laboratory-scale equipment.

During the first campaign in TRU, a few hundred millicuries of ^{131}I were released to the environment; during the second campaign, about 2 Ci were released. These quantities represented from 1 to 2% of the original inventories of ^{131}I at the times of the aluminum dissolutions. There were no provisions for sampling off-gas streams to determine DF's. The iodine was released in spurts--not continuously--and it was not possible, in most cases, to determine precisely which process steps had caused the releases. However, the releases appeared to be closely related to: (1) the transfer of caustic solutions between tanks, and (2) the Pubex process steps that involved air sparging to contact the aqueous and organic phases. The off-gas system consisted of a caustic scrubber followed by iodine-impregnated charcoal. On the basis of data collected during the campaigns, plus some laboratory data, we concluded that the charcoal traps were being poisoned by organic vapors.

Prior to the third campaign, a small, experimental off-gas system was installed. Every tank that was expected to contain a significant quantity of ^{131}I was vented through this system. The smaller (5 cfm as compared with 300 cfm for the plant) off-gas stream was treated in a caustic scrubber to remove HCl vapors. The gas was then heated electrically to 350°C, passed through a bed of oxidation catalyst (Mine Safety Applicance Company's "Hopcalite"), cooled to 130°F, passed through a 7-in.-thick bed of iodine-impregnated charcoal, and then added to the plant cubicle and vessel off-gas stream.

The iodine DF across the catalyst bed, cooler, and charcoal bed was as high as 2.8×10^4 for the first two days prior to introduction of the organic solvent into the system for the extraction of plutonium. The DF averaged 1×10^3 during the remainder of the campaign. This decrease in DF is believed to be the result of incomplete oxidation of organic materials during periods when the organic concentration was very high in the air entering the oxidizer. Prior to the introduction of organic materials into the system, the ratio of the total ^{131}I in the

plant (300 Ci) to the amount released per day from the small treatment system (0.004 mCi) was 7.5×10^7.[14]

While this performance cannot be extrapolated to a large fuel processing plant, it is encouraging to note that retention factors of 10^8 are obtainable under carefully controlled conditions in an engineering-scale operation.

Potential Iodine Removal Methods. - Iodine is present in many forms in fuel processing off-gas, for example, I_2, HI, HOI, organic iodide vapors and aerosols (particles), inorganic iodide aerosols, and various forms of iodine sorbed on organic or inorganic particles. The presence of multiple species makes iodine removal difficult. Either several removal methods must be used, or the iodine must be converted to a single form which is removed efficiently. Conversion to elemental iodine may meet this objective since several methods show peak efficiencies for removal of this reactive species.

Literature containing valuable information on treatment systems for iodine is available from the Nuclear Safety Program[15-17] and fuel processing plants. However, much of this is difficult to interpret for use in fuel processing plants. The systems developed in the Nuclear Safety Program (sprays, foams, silver surfaces, etc.) are not always applicable to those of processing plants. The latter require continuous treatment methods which can remove kilograms of iodine per day. Valuable practical information is available from units in present fuel processing plants, but it largely consists of overall DF's rather than analysis of performance by species of iodine. Thus, complete and accurate kinetic evaluation of treatment systems is not available, and scale-up and projection for new systems are impossible without further detailed data.

Source of Species. - Elemental iodine is formed directly by heating fuels or during dissolution of fuels in acid. Nitrous acid converts iodides or iodates to elemental iodine. Iodates are formed from oxidation of I_2 by nitric acid; iodides are formed by reaction of I_2 with reductants such as uranium metal. Hypoiodous acid (HOI) is formed by hydrolysis of iodine ($H_2O + I_2 \rightarrow H^+ + I^- + HOI$); thus the formation of

HOI is favored at high pH. Organic iodides are formed when organic vapors contact iodine, and also by radiolysis of organic materials in the presence of iodine. Methyl iodide has received the most study, but many other organic iodides are produced in either liquid or vapor phases. The organic components are derived from impurities in or on the fuels, solvents, degradation of solvents, impurities in air, etc. Organic iodides have been produced by heating fuel elements[18] and in the off-gases from the Purex plants at Hanford[19] and Savannah River.[20] The major constituents were methyl, butyl, hexyl, and heptyl iodides, although some octyl iodide was detected. Finally, various aerosols and particulates have been reported.[17]

Aqueous Scrubbers. - Scrubbing, which is a primary chemical engineering method for removing large masses of chemicals from off-gas streams, is the preferred primary treatment method for processing plants where kilograms of iodine (and nitrogen oxides) must be retained. Eggleton[21] made a theoretical study (i.e., calculations) of partition coefficients, which indicates that extremely high scrubber efficiencies can be obtained by proper control of variables. Assuming elemental iodine as the only volatile species, partition coefficients vary from 83 at high iodine concentrations and low pH levels to greater than 10,000 at high pH levels and low iodine concentrations. The values at 100°C are smaller than those at 25°C at high iodine concentrations but increase more rapidly as the pH is increased and the iodine concentration is decreased. The effects of oxidizing and reducing agents are extremely important, particularly with dilute iodine solutions, and could explain many of the anomalous values in the literature.

Caustic. - Caustic is the most important scrub solution because of its high potential efficiency for iodine removal and its suitability as a storage medium for the iodine (Table 4.13). Caustic scrubbers have been studied and used extensively in plant installations. Laboratory studies show that caustic removes I_2 and HI with DF's as high as 10^2 to 10^4.[9,22-24] Coops obtained a DF of greater than 10^4 in a moderately large hot-cell scrubber.[25] However, caustic scrubbers are probably inefficient for removing organic iodides, or particles. The chemistry and the performance of HOI are largely unknown. Cartan et al.[26] suggest that the volatile species from alkaline solutions is HOI or a species in equilibrium with

Table 4.13. Methods for Removal of Iodine from Off-Gas Streams

Species evaluated: I_2, HI, HOI, organic iodides (MeI,[a] etc.)

System	Performance			Comments
	Good	Poor	Unknown	
Caustic solution	I_2, HI	MeI	HOI	Partition coefficients of 10^2-10^4 dependent on pH and redox potential; reported data erratic because of presence of alkyl iodides or particulates.
Caustic solution with additives (thiosulfate for reduction or amines for fixation)	I_2, HI		MeI HOI	Conflicting data on efficacy of additives; probably affected by presence of other species.
Dilute nitric acid	I_2, HI		MeI HOI	Economically attractive for nitric acid recycle; efficient at low I_2 concentrations.
Mercuric nitrate--nitric acid	I_2, HI, MeI		HOI	Improves DF and loading attainable with nitric acid; equilibrium data are needed.
Activated charcoal	I_2, HI	MeI	HOI	I_2 capacity not seriously affected by organic matter, or carbon dioxide; fire hazard with nitrogen oxides.
Impregnated charcoal (KI)	I_2, HI, MeI		HOI	MeI (but not I_2) capacity seriously affected by organic, water, or carbon dioxide vapors; fire hazard with nitrogen oxides.
Impregnated charcoal (triethylenediamine)	I_2, HI, MeI		HOI	MeI (but not I_2) capacity affected by CO_2, but not seriously affected by organics or water; combination with KI-impregnated charcoal attractive; triethylenediamine released at approximately 190°C.
Silver reactors (185°C)	I_2, HI		MeI, HOI	Cost of silver may be excessive for primary unit; requires high temp.; operating problems in maintaining temp. and in regeneration.
Hydrous zirconium oxide (saturated with iodine)	I_2, HI	MeI	HOI	Zirconium oxide is inexpensive; good for I_2; does not burn.
Impregnated zeolite (silver)	I_2, HI, MeI		HOI	Cost of silver may be excessive for primary unit; little operating experience; does not burn; good at high temperature.
Ion-exchange resins	I_2, HI		MeI, HOI	Possibly uneconomical; operating problems in maintaining wet surfaces.
Catalytic oxidizer (Hopcalite catalyst, O_2, 150-500°C, charcoal bed)	I_2, HI, MeI		HOI	Converts organic iodides to CO_2, H_2O, I_2; requires high temperature.

[a]MeI = methyl iodide.

it. They found that the ratio of the volatile species to elemental iodine was greater than 10 in the gas sparged from water at pH 10.2 (conditions which favor the formation of HOI). The vapor pressure of HOI has not been measured, but calculations indicate that it should be less than that of iodine.[21] The reasons for the relatively poor performance of large scrubbers at Idaho[12] and Hanford[6] (i.e. ~70% and 90% removal, respectively) are unknown. The low concentrations of iodine in the gas phase, that is, 1×10^{-3} to 80×10^{-3} μg I_2/ft^3 could be a factor at Hanford. (Concentrations have not been reported at Idaho.) Tests of the caustic scrubber[18,19] for the Oak Ridge Research Reactor showed a removal efficiency of 99%, with an inlet iodine concentration of 15 μg/ft^3. Other factors are the unknown amounts of particulates and organic iodides present. No reports have been found in the literature which relate the efficiencies or kinetics of caustic scrubbing to irradiation rate, concentration of iodine in both phases, concentration of caustic, temperature, and contacting efficiency.

The use of additives with caustic to reduce[9,19,17] iodine and iodate to iodide increases the removal efficiency for iodine. Thiosulfate is a commonly used reductant. The removal efficiency of the Idaho system increased from 90 to 97% when the caustic contained thiosulfate.

Nitric Acid and Mercuric Nitrate--Nitric Acid. - Nitric acid and mercuric nitrate--nitric acid are effective scrubbing media for removing both iodine and nitrogen oxides from gas streams. Stromatt[27] studied this system as a function of the concentrations of HNO_3, $Hg(NO_3)_2$, and I_2; flow rate; and temperature. Iodine removals greater than 99.9% were obtained at iodine gas-phase concentrations ($<7 \times 10^{-9}$ g/ml) with either 5% HNO_3 or 5% HNO_3 that was 10^{-4} M in $Hg(NO_3)_2$. With higher iodine concentrations (1×10^{-7} g/ml), the retention in the pure acid system dropped to 50% , whereas with a $Hg(NO_3)_2$ - HNO_3 system (10^{-2} M Hg) the removal remained at greater than 99.9%. Crouse et al.[28] have shown that methyl iodide and alkyl iodides as high as 2-octyl iodide react rapidly (dodecyl reacts slowly) with the $Hg(NO_3)_2$- HNO_3 scrubbing solution. Removal efficiencies of greater than 99.9% were obtained for methyl iodide.

Silver Reactors. - Silver reactors have been used routinely at Hanford and Savannah River for many years. Iodine removal varies from 99 to 99.9% in plant operations.[6] Temperatures of 375 to 400°F are required for efficient operation. At higher temperatures (413°F) the silver nitrate coating melts and runs off, while at temperatures below the condensation point of nitric acid (~230°F) the silver nitrate coating dissolves. In the absence of nitrogen oxide fumes, the silver nitrate decomposes to Ag_2O, and temperatures above 413°C could be used. Research is now in progress to define the efficiency for removing iodine from organic compounds.[5]

Solid Adsorbents. - Solid adsorbents are useful as final polishing agents in instances where high removal is required but the mass to be sorbed is low.

Charcoal and Impregnated Charcoal Beds. - Charcoal and impregnated charcoal beds have been studied extensively by Adams, Ackley et al.[15,29-32] Decontamination factors up to 10^5 can be expected for elemental iodine on plain or impregnated charcoal. Water vapor, organic compounds (not iodides), and carbon dioxide do not reduce the adsorption capacity of elemental iodine seriously. If alkyl iodides are formed by the reaction of organic vapors and iodine, their sorption will be poor on plain charcoal under humid conditions. Oxidizing gases are hazardous. Temperatures must be less than about 150°C to prevent desorption of iodine from iodized charcoal. Maximum adsorption of iodine will be obtained at low (room) temperatures in a nitrogen stream with low relative humidity after removal of the other constituents. These laboratory tests have shown that the elemental iodine content of air can be reduced, by use of charcoal, from 0.1 mg/m^3 to 0.00001 mg/m^3.[33] Methyl iodide can be removed by factors of 10^3 to 10^4 by charcoals impregnated with $KI-I_2$ or triethylenediamine.[31,34] However, acid vapors, organic vapors, and moisture adversely affect the efficiency of the impregnated charcoal. Charcoal impregnated with $KI-I_2$ and containing a high loading of iodine starts to lose iodine at about 150°C.[31] Plain charcoal with a lower iodine loading tends to lose iodine at about 250°C or higher, depending on the type of charcoal, iodine loading, etc. Charcoal impregnated with triethylenediamine also loses iodine at elevated temperatures, and temperature surges have been observed at approximately 190°C.[31,34] Most

of the work with impregnated charcoals (effects of gas velocity, temperature, etc.) has been confined to gas concentrations of about 17 mg CH_3I/m^3; in a few experiments, the concentration was about 2 mg/m^3. Plain charcoals have been studied extensively to determine iodine adsorption as a function of loading, gas concentration, temperature, velocity, and effect of competing gases. The capacity of iodized charcoal (2 to 5% $KI-I_2$) for adsorbing elemental iodine is not seriously affected by the large loading of iodine present initially. Adams and Ackley have adsorbed up to 60 mg of I_2 per gram of adsorbent, with observed removal of 99.9^+%.[32]

Zeolites Impregnated with Silver. - Zeolites impregnated with silver have been studied by Maeck et al.[35] The silver form of the zeolite is prepared by treating the zeolite with silver solution, washing to remove excess silver nitrate, and drying. Linde molecular sieve 13X impregnated with silver removes iodine and methyl iodide by DF's greater than 1×10^4. This new material is very promising because it (1) does not burn, (2) adsorbs iodine at temperatures up to 500°C, (3) retains iodine up to 1000°C, and (4) adsorbs iodine rapidly.

Alumina-Bismuth Metal Mixtures or Iodized Hydrous Zirconium Oxides. - Alumina-bismuth metal mixtures or iodized hydrous zirconium oxides showed high efficiencies ($>10^4$) for removing elemental iodine but not methyl iodide.[35] These materials may be cheaper than zeolites impregnated with silver.

Ion Exchange Resins. - Dowex 50 showed a 98% removal of iodine but poor removal of methyl iodide.[35] Amberlite-IRA-400-OH showed a high efficiency for removal of iodine (DF of 10^3) but was not tested with methyl iodide.[36]

Catalytic Oxidation and Adsorption. - Catalytic oxidation followed by adsorption on charcoal has been studied by Parker,[37] Adams,[38] and Cathers.[39] A Hopcalite bed ($CuO-MnO_2$ catalyst) will oxidize organic compounds and convert organic iodides to H_2O, CO_2, and I_2 at 150 to 500°C. Similar catalysts, such as Al_2O_3 or CuO, can probably also be used. The elemental iodine can be removed in charcoal, silver zeolite, or

other beds listed above by very high factors (i.e., 10^3 to 10^5). The cost of this oxidation step should not be excessive. In the present plants, it is necessary to heat the off-gases to 380°F (183°C) before they enter the silver reactor treatment system, and subsequently cool them.

Particulate Removal. - The removal of particulates from process plant off-gas is an important and necessary step in controlling iodine release to the atmosphere. The development, in recent years, of high-efficiency particulate air (HEPA) filters ensures the removal of up to 99.97% of 0.3-μ particles under optimum conditions.[40] Whether this degree of retention will be adequate to meet atmospheric emission codes in future years will depend on the amount of iodine activity associated with particles in the 0.01- to 0.1-μ range. The importance of this range of particle size is now being studied.[41]

4.10.3 Noble Gases

Several processes have been, or are being, developed to isolate the noble gases from off-gas streams.[42,43] This technology has advanced to the point that methods are now available to isolate the noble gases, including the ^{133}Xe from 30-day-cooled LMFBR fuel, provided they are discharged from the process in a small off-gas stream. Selection of the particular design to optimize the process must await information concerning the volume and composition of the off-gas.

The processes studied include: (1) adsorption on charcoal or molecular sieves at ambient temperatures, (2) adsorption on charcoal or molecular sieves at low temperatures, (3) separation by cryogenic distillation, (4) adsorption in hydrocarbons or chlorofluoromethanes, (5) separation by perm-selective membranes, and (6) precipitation with clathrates or as fluorides. These have been developed to various degrees (Table 4.14). Another alternative which has been considered is the injection of the gases into porous underground media.[44]

The adsorption of noble gases on charcoal or molecular sieves at ambient temperature is the process which has had the most extensive use.[45]

Table 4.14. Processes for Removal of ^{85}Kr from Off-Gas at a Fuel Processing Plant[a]

Process	% Kr[b] Removal	Development Status	Comments
Adsorption on room-temperature charcoal beds or molecular sieves	99	Pilot plant with reactors	Simple operation; charcoal can ignite.
Adsorption on low-temperature charcoal or silica gel beds	99	Pilot plant with processing plant	Small bed volumes; charcoal can ignite.
Cryogenic distillation	99	Pilot plant test with processing plant	Small equipment; ozone explosion hazard.
Absorption in hydrocarbons or chlorofluoromethanes	99	Nonradioactive pilot plant tests	Small equipment; no fire hazard.
Separation by permselective membranes	99	Laboratory studies; large-scale single unit constructed	Small equipment; no fire hazard.
Precipitation with clathrates or as fluorides	Unknown	Laboratory studies	^{85}Kr collected as solid; degraded by radiation; slow precipitation at 150 atm.

[a]Varying gas pretreatments required.

[b]Minimum design values.

This is a simple method for delaying the release of the noble gases to the atmosphere to allow the short-lived xenon to decay. It has disadvantages in that (1) large beds of charcoal are required, (2) the charcoal is subject to fire, and molecular sieves are subject to explosion due to local heating by radioactive particles, and (3) the ^{85}Kr is released to the atmosphere instead of being concentrated for permanent storage. The fire and explosion hazards may be eliminated or minimized by suitable design or operational procedures.

The adsorption on charcoal or molecular sieves at low temperatures[46] has the advantage of decreasing the size of the adsorption bed, but the fire hazard remains. Further disadvantages are the cost of cooling and explosion hazards due to the higher concentration of ozone produced by the radiation of oxygen. Materials which would freeze or condense must be completely removed prior to injection into the bed to prevent the equipment from plugging. Nitrous oxide gas has been particularly troublesome in this system and must be destroyed by contact with hydrogen and a palladium catalyst.

In the cryogenic separation, the noble gases and part of the air are liquefied. The noble gases are then separated and concentrated by fractional distillation.[46,47] This process has been demonstrated at a gas flow rate equal to the off-gas from a 3-ton/day fuel processing plant but at about one-fifth the activity level. As with all of the low-temperature operations, the condensable gases must be removed to a very low level prior to the noble gas treatment. Besides the physical difficulties with solids in the system, there are explosion hazards from liquid ozone, etc., if the head-end removal process is not effective.

The study of the separation of noble gases from air by the adsorption in chlorofluoromethanes (Freon) or hydrocarbons has been developed to the nonradioactive pilot plant stage.[48] This is a very versatile system. Since the operating temperature is higher than the low-temperature charcoal bed adsorption or cryogenic distillation method (~ -20°F vs -290°F), the level to which the condensable gases must be removed is less stringent. Operation at or near ambient temperatures seems to be a possibility and is being investigated. Degradation of the solvent by irradiation is not expected to be limiting but has not been investigated.

The removal of noble gases by use of permselective membranes has been investigated on a laboratory scale.[49] The gas permeabilities and separation factors are sufficiently large to make the commercial application attractive, but engineering development is required. Since this equipment is small and operates at ambient temperatures, it avoids the difficulties with condensable gases and the explosion hazards of the low-temperature processes. This process is also versatile and is attractive because it is easily started and stopped. The membrane has high radiation stability, but its life under operating conditions has not been determined.

Precipitation of the noble gases with clathrates or as fluoride compounds has been proposed. This process would yield solids, which would simplify permanent storage. It does not appear to be economically attractive as a primary separation method due to the very low concentration of noble gas in the off-gas streams, but may be attractive for solidifying concentrated gases for storage. Other methods for fixing the gases in solids for permanent storage are discussed in Sect. 4.9.

It can be seen that there are several processes which may be used to remove noble gases from the off-gas of power reactor fuel processing plants. The most economical process will depend upon the volume of the off-gas and the degree of removal of noble gas which will be required at a particular processing plant. The recovered noble gases must be stored in conventional pressurized gas cylinders until advanced disposal methods are available (see Sect. 4.9).

4.10.4 Tritium

As was stated in the introduction, tritium could be disposed of by volatilizing it up the stack. However, if tritium is contained in a small off-gas stream, such as the voloxidizer off-gas, it would be possible to fix it on a drying agent, such as a molecular sieve. This is not a crucial problem and should receive attention only after the off-gas system has been established by the overriding iodine problem.

4.11 Radiation and Shielding Studies - Task 11

The successful development of economical methods for processing irradiated LMFBR fuels requires detailed information with respect to the concentrations of radionuclides, sources and energy distribution of radiation, thermal power, and shielding requirements. Concentrations of elements and radionuclides not normally considered in reactor design and evaluation studies (Table 4.15) must be known to properly evaluate separations techniques, waste disposal schemes, and plant containment and siting features. The thermal power and the distribution of radiant energy are needed as a function of postirradiation decay time for determination of cooling requirements and solvent degradation studies. Shielding requirements dictate feasible engineering concepts throughout all of the fuel recycle.

The following sections will summarize current methods for estimation of the radiation properties of irradiated LMFBR fuels and present applications of these data to aqueous processing.

4.11.1 Nuclear Data

Principal references and a tabulation of the nuclear data that are currently used in the ORIGEN program for calculating properties of irradiated LMFBR fuels are presented in Appendix D of this report. This tabulation includes cross section data averaged over the LMFBR core spectrum shown in Fig. 4.15. The assumed spectrum is a smoothed version of that computed by Kusters and Metzenroth[1] for a typical LMFBR core composition. It agrees well with core-averaged spectra computed by General Electric for alternative 1000-Mw(electrical) LMFBR follow-on designs.[2] The spectrum is generally somewhat softer in LMFBR axial blankets, but the assumed spectrum has been found to be adequate for estimating purposes since the derived cross sections provide a reasonable estimate of plutonium production in LMFBR blankets. Core spectra could be appreciably softer in LMFBR's that use beryllium as a moderator to achieve a negative sodium void coefficient, but a hard spectrum with appropriate engineered safety features is currently favored because of economic considerations.

Table 4.15. Some Important Radionuclides Present in Spent LMFBR Fuel

Nuclides	Reasons for Importance in Aqueous Fuel Processing
^{3}H, ^{85}Kr, ^{133}Xe, ^{131}I, ^{129}I	Volatile materials that may potentially be released to the environment.
^{103}Ru, ^{106}Ru, ^{127m}Te, ^{129m}Te, ^{137}Cs	Semivolatile materials that may potentially be released to the environment.
^{89}Sr, ^{91}Y, ^{95}Zr, ^{95}Nb, ^{103}Ru, ^{106}Ru, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Pr	Sources of heat, penetrating radiation, and radiation damage
^{90}Sr, ^{137}Cs, ^{241}Am, ^{244}Cm	Hazardous materials in aged waste.
^{236}Pu, ^{238}Pu, ^{240}Pu, ^{241}Am, ^{237}U, ^{232}U	Sources of penetrating radiation in recycled plutonium.
^{238}Pu, ^{242}Cm, ^{244}Cm	Sources of neutron radiation in spent fuel shipment.

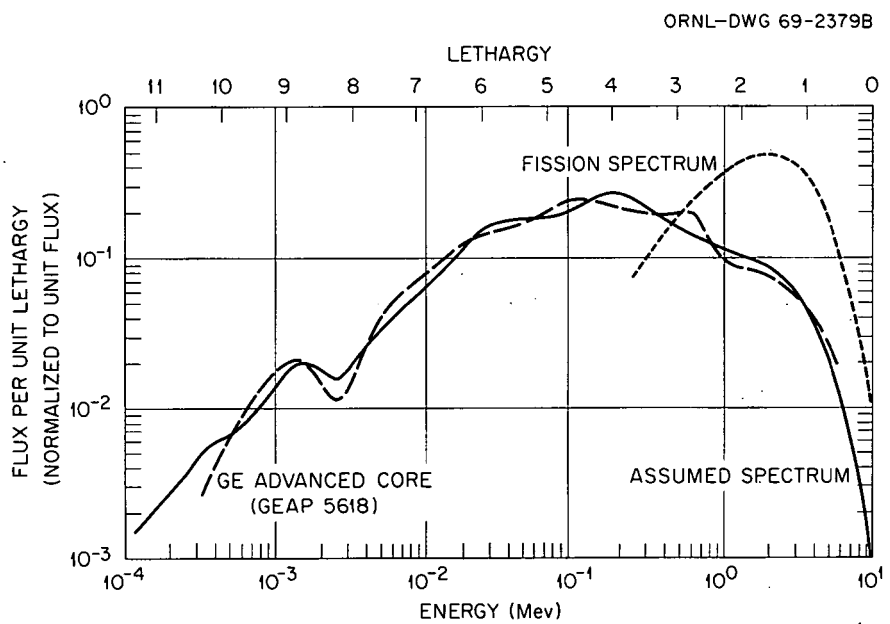

Fig. 4.15. Comparison of the Assumed LMFBR Core Flux Spectrum with the Fission Spectrum and the Spectrum Computed for the General Electric Follow-On Advanced Design.

The prediction of heat generation properties of individual fission products has often been unsatisfactory because of uncertainties in average energy of negatron and positron reactions. Average beta energies for the current compilation were calculated with the computer program SPECTRA,[3] based on data and/or estimates of the maximum negatron or positron energy and the degree of forbiddeness of the transition. Typical data presenting the calculated average-to-maximum energy for allowed and first-forbidden transitions as a function of maximum energy and atomic number are shown in Fig. 4.16.

4.11.2 Reactor Irradiation Studies

The computer programs that have been developed at ORNL for calculation of the transient masses, concentrations, radioactivity, and thermal power of radionuclides resulting from irradiation are PHOEBE and ORIGEN. The PHOEBE program[4] computes gross fission product thermal properties and a 12-group photon source as a function of postirradiation decay time for a specified specific power history. PHOEBE employs correlations of gross data that are available for ^{235}U thermal fission products and calculated data for fast and thermal fission in ^{239}Pu. The 12-group gamma source calculated by PHOEBE is generally satisfactory for estimating shielding requirements for spent LMFBR fuels. The PHOEBE results can be made progressively more accurate for LMFBR fuels with the addition of correlated data generated by such programs as ORIGEN.

The ORIGEN program,[5] which was used to produce the results shown in Sect. 3.2, is a versatile program for solving extremely large systems of ordinary linear differential equations by a series technique. Nonlinear effects are treated either by a Taylor series expansion or a programmed step function. It is presently the preferred program for calculating transient concentrations, activities, and thermal power of the approximately 700 important nuclides (actinides, fission products, and light-element activation products) in irradiated LMFBR fuel. The allowable types of nuclear transmutation include negatron, positron, and alpha decay; isomeric transition; and a variety of neutron reactions (n-capture; n-fission; n,2n; n,3n; n,α; and n,p). The program presently requires one-group cross

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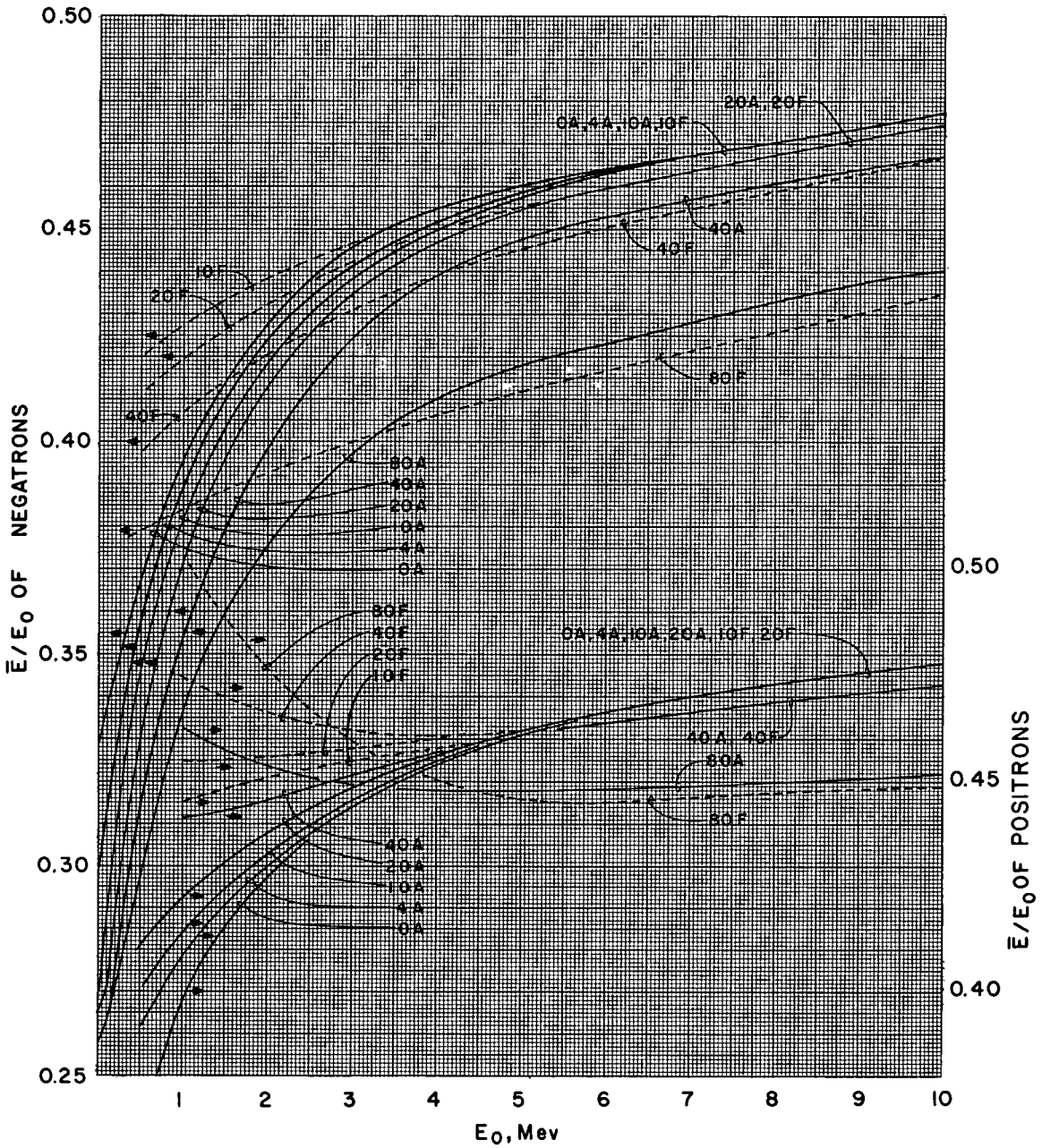

Fig. 4.16. Average Energy, $\bar{E}$, of Negatron and Positron Spectra as a Function of Maximum Energy, E_0, and Proton Number for Allowed (A) and First-Forbidden (F) Transitions.

sections averaged over the reactor spectrum of interest. Input data consist of initial concentrations and either neutron flux or thermal power in 10 or more time steps.

4.11.3 Radiation Dose Rates and Shield Attenuation Kerma

The radiation and shielding data of primary concern in aqueous processing of LMFBR fuels fall into two distinct categories: those associated with the fission products of spent fuel and those associated with the handling of the plutonium product. Shielding calculations for the fission product gamma sources are relatively straightforward after the gamma energy spectrum has been calculated by a suitable program such as PHOEBE. Accurate assessment of the dose rates and shielding requirements for the neutron sources in plutonium product requires the use of techniques that have been developed only in the last several years.[6]

Radiation Source Data. - Typical photon release rate and spectrum data for LMFBR fuel as a function of postirradiation time are presented in Fig. 4.17. These data were calculated with the PHOEBE program, assuming irradiation to a burnup of 100,000 Mwd/metric ton at an average specific power of 125 Mw/metric ton. At short decay times, shielding requirements are determined primarily by the 1.55-Mev gamma group that originates predominantly from the decay of ^{140}Ba-^{140}La. At longer decay times, shielding requirements are determined primarily by the 2.38-Mev group, to which a primary contributor is ^{144}Ce-^{144}Pr.

The estimated primary neutron source strength of LMFBR fuel is compared with that of LWR fuel in Table 4.16. The isotopic content of those nuclides which constitute important neutron sources were calculated with the ORIGEN code. Most of the neutrons originate from spontaneous fission of the isotopes ^{242}Cm and ^{244}Cm. The total neutron source strength of an array of fuel elements will be higher than the source strengths presented in Table 4.16 by perhaps a factor of 3 to 20 because of the effect of subcritical neutron multiplication. Assuming arrays of equal neutron multiplication factor, these results indicate that the neutron source of spent LMFBR core fuel will be greater than the neutron source of enriched uranium LWR fuel but less than that of plutonium recycle LWR fuel.

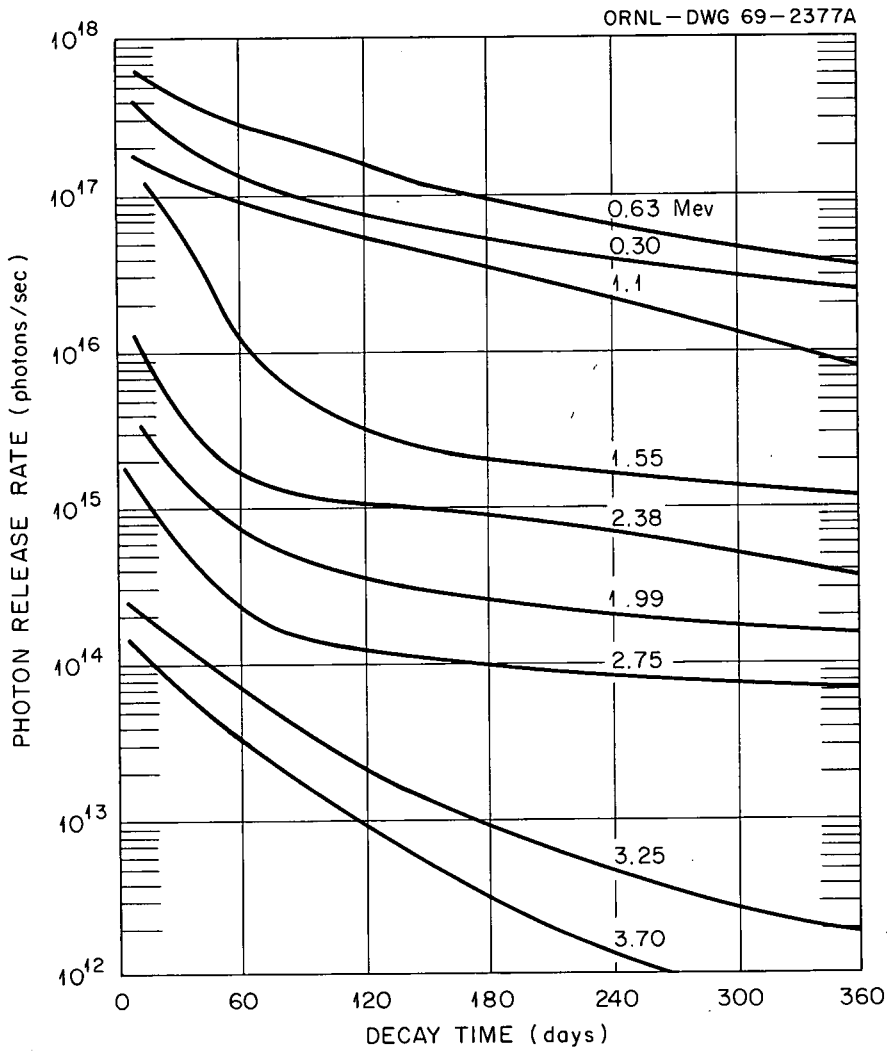

Fig. 4.17. Effect of Postirradiation Decay Time on the Photon Release Rate of Fission Products Resulting from the Irradiation of 1 Metric Ton of LMFBR Fuel to 100,000 Mwd/metric ton at an Average Specific Power of 125 Mw/metric ton.

Table 4.16. Estimated Neutron Source Strengths of Spent Power Reactor Fuels
(Does not include neutrons from subcritical multiplication)

Isotope	LWR[a]			LWR-Pu Recycle[a]			LMFBR[b]		
	Isotope Content[c] (g/ton)	Neutron Source (α,n) (sec^{-1} ton^{-1} x 10^{-6})	Neutron Source SF[d] (sec^{-1} ton^{-1} x 10^{-6})	Isotope Content[c] (g/ton)	Neutron Source (α,n) (sec^{-1} ton^{-1} x 10^{-6})	Neutron Source SF[d] (sec^{-1} ton^{-1} x 10^{-6})	Isotope Content[c] (g/ton)	Neutron Source (α,n) (sec^{-1} ton^{-1} x 10^{-6})	Neutron Source SF[d] (sec^{-1} ton^{-1} x 10^{-6})
^{238}Pu	160	5.0	0.4	45	1.4	0.1	1,800	55.6	4.8
^{239}Pu	5400	0.4	0.0	1500	0.1	0.0	120,000	8.6	0.0
^{240}Pu	2200	0.6	2.0	1000	0.3	0.9	52,000	14.	49.
^{241}Am	62	0.4	0.0	60	0.4	0.0	1,400	8.2	0.0
^{242}Cm	4.4	52.9	87.7	35	418.	693.	48	574.	952.
^{244}Cm	30	7.3	326.	510	124.	5560.	42	10.2	459.
Total		66.	416		544.	6250		670.	1460.
Overall total		480			6800			2130	

[a]Fuel had been irradiated to a burnup of 33,000 Mwd/metric ton and allowed to cool for 150 days.

[b]Fuel had been irradiated to a burnup of 100,000 Mwd/metric ton and allowed to cool for 60 days.

[c]"Ton" in this table refers to a metric ton of fuel as charged to the reactor.

[d]SF = spontaneous fission.

Based on experimental data from LWR's and estimated cross sections in LMFBR's, the neutrons produced by spontaneous fission and (α,n) reactions constitute the important source of penetrating radiation in plutonium product (and recycle plutonium feed); by comparison, the primary gamma rays of high yield have relatively low energy and are easily shielded. The transient radioactivity of the decay products of plutonium having the isotopic composition of LWR product are shown in Table 4.17. These plutonium composition data were calculated with the ORIGEN program, assuming a fuel burnup of 33,000 Mwd/metric ton at a specific power of 30 Mw/metric ton in the Diablo Canyon reactor; however, they are typical of experimental data.[7] The predominant source of radiation from lightly shielded plutonium of this composition is the 0.06-Mev gamma ray that occurs in nearly 100% of the disintegrations of ^{241}Am. This gamma radiation is effectively shielded by 1/8 to 1/4 in. of lead. The hard gamma radiation (predominantly the 2.62- and 0.58-Mev gamma radiation that occurs in 100% of the disintegrations of ^{208}Tl) resulting from ^{236}Pu daughters begins to give dose rates comparable to those from the neutron sources after plutonium post-purification times of greater than one year. Since plutonium is generally stored as a nitrate solution, a relatively simple ion exchange treatment for removing most of the ^{236}Pu daughter activity may be employed, if necessary, before the fuel is shipped or fabricated.

Estimates of the isotopic composition and neutron source strength of fuel core charges for LMFBR's and LWR's are shown in Table 4.18. These data reflect current estimates that the plutonium recycled from LMFBR's will contain less ^{236}Pu and ^{238}Pu than plutonium recycled from LWR's. The neutron source strength per unit of each plutonium isotope is based on relatively abundant experimental information.[8-10] The two primary sources of neutrons are: those from (α,n) reactions of ^{239}Pu and oxygen, and spontaneous fission of ^{240}Pu.

Shield Attenuation Kerma. - The energy spectrum of neutrons from the (α,n) reactions of ^{238}Pu and oxygen[11,12] is compared with the thermal-neutron fission spectrum (the approximate spectrum of neutrons from spontaneous fission) of ^{235}U in Fig. 4.18. Calculated neutron and secondary

Table 4.17. Isotopic Activity of Plutonium Product from a Light Water Reactor as a Function of Cooling Time

Radioactivity expressed as curies per metric ton of fuel charged to reactor.

Nuclide	Cooling Time (days)					
	0	3	30	300	3,000	30,000
^{208}Tl	0.0	1.45×10^{-8}	1.39×10^{-4}	1.91×10^{-2}	5.05×10^{-1}	3.58×10^{-1}
^{212}Pb	0.0	4.03×10^{-8}	3.86×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{212}Bi	0.0	4.03×10^{-8}	3.86×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{212}Po	0.0	2.58×10^{-8}	2.47×10^{-4}	3.39×10^{-2}	8.97×10^{-1}	6.36×10^{-1}
^{216}Po	0.0	1.04×10^{-6}	4.39×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{220}Rn	0.0	1.04×10^{-6}	4.39×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{224}Ra	0.0	1.04×10^{-6}	4.39×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{228}Th	0.0	6.26×10^{-6}	6.16×10^{-4}	5.30×10^{-2}	1.40	9.94×10^{-1}
^{232}U	0.0	4.20×10^{-3}	4.16×10^{-2}	3.80×10^{-1}	1.73	9.94×10^{-1}
^{234}U	0.0	4.67×10^{-3}	4.67×10^{-2}	4.66×10^{-1}	4.52	3.45×10^{1}
^{236}U	0.0	1.24×10^{-5}	1.24×10^{-4}	1.24×10^{-3}	1.24×10^{-2}	1.24×10^{-1}
^{237}Np	0.0	2.25×10^{-7}	2.24×10^{-5}	2.21×10^{-3}	1.94×10^{-1}	7.55
^{236}Pu	5.32×10^{1}	5.31×10^{1}	5.21×10^{1}	4.35×10^{1}	7.21	1.12×10^{-7}
^{238}Pu	2.03×10^{5}	2.03×10^{5}	2.02×10^{5}	2.01×10^{5}	1.90×10^{4}	1.07×10^{5}
^{239}Pu	3.63×10^{4}	3.63×10^{4}	3.63×10^{4}	3.63×10^{4}	3.63×10^{4}	3.62×10^{4}
^{240}Pu	5.22×10^{4}	5.22×10^{4}	5.22×10^{4}	5.22×10^{4}	5.22×10^{4}	5.18×10^{4}
^{241}Pu	1.36×10^{7}	1.36×10^{7}	1.35×10^{7}	1.30×10^{7}	8.77×10^{6}	1.70×10^{5}
^{242}Pu	1.56×10^{2}	1.56×10^{2}	1.56×10^{2}	1.56×10^{2}	1.56×10^{2}	1.56×10^{2}
^{241}Am	0.0	1.69×10^{2}	1.68×10^{3}	1.65×10^{4}	1.36×10^{5}	3.46×10^{5}
Total	1.39×10^{7}	1.39×10^{7}	1.38×10^{7}	1.33×10^{7}	9.18×10^{6}	7.11×10^{5}

Table 4.18. Estimates of the Isotopic Content and Neutron Source Strength of PuO_2 Recycle Fuels

	Isotope							Total
	^{238}U	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	
LMFBR Core[a]								
Isotope content, g/kg (U + Pu)	761.6	1.4×10^{-6}	1.7	163.4	50.8	13.9	8.5	
wt % in Pu		0.6×10^{-6}	0.72	68.57	21.30	5.83	3.58	
Neutron source, neutrons sec^{-1} $[kg\ (U + Pu)]^{-1}$								135,000
From SF			4,500		48,000		14,000	66,000
From $(\alpha,\underline{n})$			42,000	11,000	12,000	3,500	33	69,000
LMFBR Core[b]								
Isotope content, g/kg (U + Pu)	781.3	23×10^{-6}	2.6	129.5	51.8	26.0	8.76	
wt % in Pu		11×10^{-6}	1.2	59.2	23.7	11.9	4.0	
Neutron source, neutrons sec^{-1} $[kg\ (U + Pu)]^{-1}$								164,000
From SF			6,900		49,000		15,000	71,000
From $(\alpha,\underline{n})$			65,000	8,700	13,000	6,600[c]	34	93,000
LWR Pu Recycle Core[b]								
Isotope content, g/kg (U + Pu)	956.4	4.6×10^{-6}	0.52	25.8	10.3	5.2	1.7	
wt % in Pu		11×10^{-6}	1.2	59.2	23.7	11.9	4.0	
Neutron source, neutrons sec^{-1} $[kg\ (U + Pu)]^{-1}$								33,000
From SF			1,400		9,800		2,900	14,000
From $(\alpha,\underline{n})$			13,000	1,700	2,500	1,300[c]	6.6	19,000

[a]Mixed plutonium from LMFBR core-blanket recycle.

[b]Plutonium from LWR recycle.

[c]From ^{241}Am, assuming one year of post-separation decay.

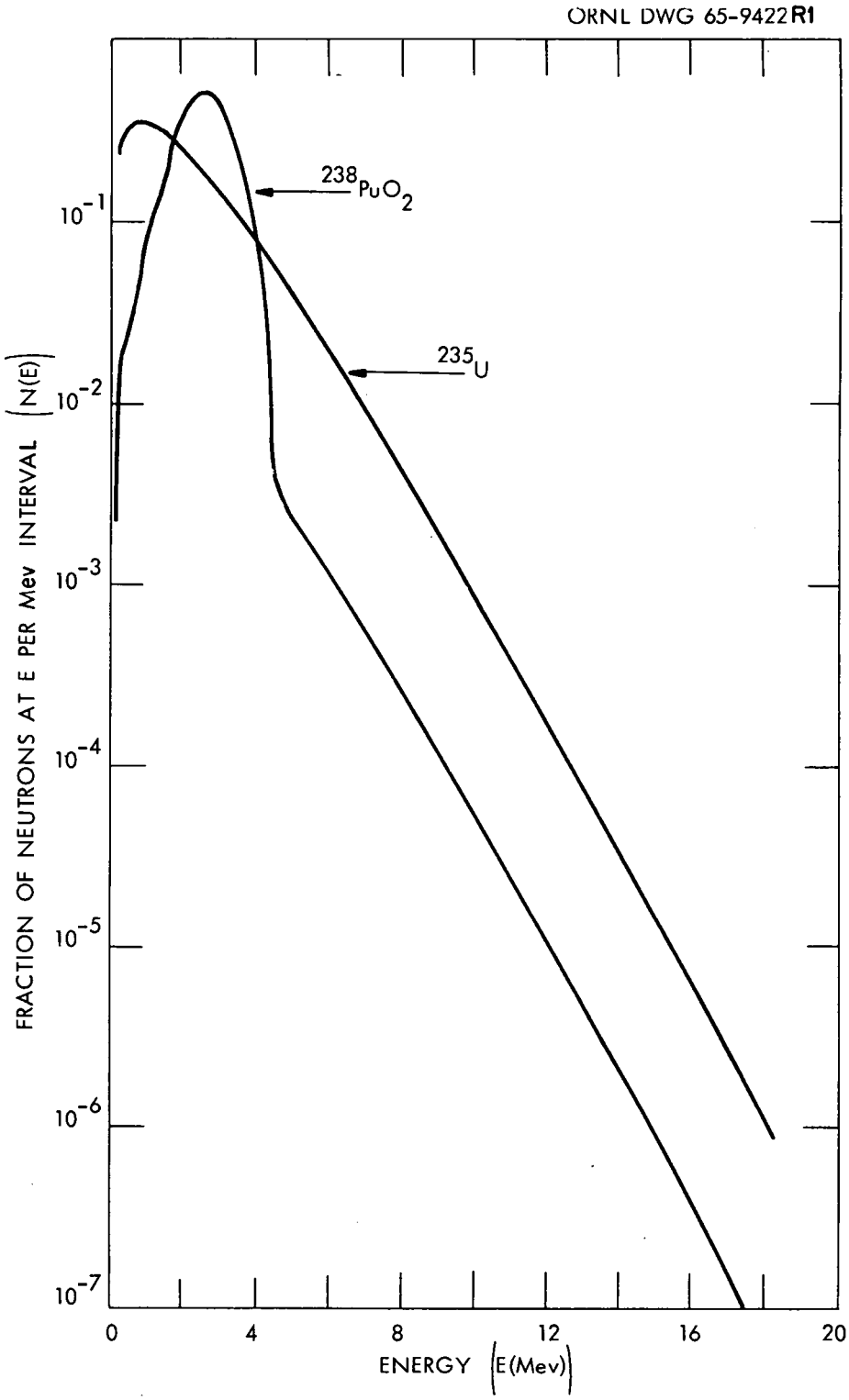

Fig. 4.18. Spectrum of Neutrons from Thermal Fission of ^{235}U and from Spontaneous Fission plus (α, n) Reactions in $^{238}PuO_2$.

gamma-ray attenuation kerma for various materials of potential interest are available in the literature.[13-15] The attenuation kerma for neutrons and capture gamma rays in conceptual LMFBR spent fuel shipping casks have also been presented in a recent report.[16]

4.11.4 Engineering Applications of Shielding and Radiation Data

Shield Requirements for LMFBR Spent Fuel Shipping Casks. - Calculations were made using the SDC code[17] to determine representative shielding requirements for a 36-element spent fuel shipping cask for LMFBR core plus axial blanket fuel (Table 4.19). The calculations were made as a function of position above the midplane of the active portion of the element. Since the cask is used to ship entire assemblies, including the nonfueled portions (which constitute approximately two-thirds of the total length), less shielding is required for the upper and lower portions and the cask is tapered to eliminate excessive shield weight.

The estimated dose rate from primary neutrons at 8 ft from the surface of LMFBR spent fuel shipping casks is compared with that from LWR fuel shipping casks in Table 4.20. These estimates indicate that supplementary hydrogeneous shields on the outside of the cask may be required for both types of fuel. Subcritical neutron multiplication and capture gamma radiation will effectively increase the requirements for neutron attenuation.[16,18] The design of neutron shields for LMFBR casks will benefit from the necessarily forthcoming designs of high-burnup LWR fuel shipping casks. Results will probably not be directly compatible, however, since it is proposed that the LMFBR cask be shielded with iron instead of lead.

Shield Requirements for Spent Fuel Processing Cells. - Figure 4.19 presents results of SDC calculations, which permit estimation of the required thickness of normal concrete cell walls for LMFBR fuel processing plants. The higher burnup and specific power of LMFBR fuel necessitate the use of 1/2 to 1 ft of additional shielding as compared with a similar mass of LWR fuel at the same postirradiation decay time.

Table 4.19. Cask Shield Thicknesses for 36 Atomics International Reference Oxide Core - Axial Blanket Elements

Cask has inside diameter of 53 in. Gamma shield requirement is based on a dose rate of 10 mrads/hr at 8 ft from the surface of the cask.

Height Above Midplane (cm)	Thickness of Iron Required (cm)			Thickness of Uranium Required (cm)		
	Decay Time (days)			Decay Time (days)		
	20	60	150	20	60	150
0	52.5	47.9	44.2	17.6	16.1	14.8
10	52.3	47.9	44.2	17.6	16.1	14.8
20	52.3	47.9	44.2	17.6	16.1	14.8
30	52.3	47.9	44.2	17.6	16.1	14.8
40	52.3	47.9	44.2	17.6	16.1	14.8
51	52.2	47.9	44.2	17.6	16.1	14.8
61	52.1	47.8	44.1	17.6	16.1	14.8
66	52.0	47.6	43.9	17.6	16.1	14.8
71	51.8	47.4	43.8	17.6	16.0	14.8
76	51.5	47.2	43.5	17.6	16.0	14.7
81	51.1	46.8	43.2	17.4	15.9	14.6
86	50.7	46.3	42.7	17.2	15.6	14.4
91	50.0	45.7	42.0	16.9	15.3	14.0
96	49.2	44.8	41.2	16.4	14.8	13.5
101.5	48.2	43.8	40.1	15.6	14.1	12.8
106.5	47.0	42.6	38.9	14.8	13.2	11.9
111.5	45.7	41.2	37.4	13.7	12.2	11.0
121.5	42.3	37.7	33.8	11.4	10.0	8.8
131.5	37.9	33.2	29.2	9.1	7.8	6.8
141.5	32.8	28.0	24.0	8.5	7.2	6.2
151.5	27.1	22.5	19.0			
168.5 and above	24.9	21.6	19.0			

Table 4.20. Estimated Neutron Dose Rates at a Distance of 8 ft from the Surfaces of LWR and LMFBR Shipping Casks

The combined neutron and gamma dose rate should be less than 10 mrem/hr for exclusive-use shipments.

Fuel	Inside Diameter (ft)	Active Length (ft)	No. of Elements	Shield	Types of Supplementary Shield		Neutron Dose Rate at 8 ft (mrem/hr)
					Inside	Outside	
LWR	4	12	4 PWR or 10 BWR	8 in. Pb + 2.5 in. Fe	Air or Al	None	18
					Water	None	0.6
					Air or Al	2 in. water[a]	5.0
LWR-Pu recycle	4	12	4 PWR or 10 BWR	8 in. Pb + 2.5 in. Fe	Air or Al	None	250
					Water	None	8.0
					Air or Al	6 in. water + 2 in. Fe[b]	5.0
LMFBR core	1.8	4	36	20 in. Fe	Sodium	None	20
					Sodium	2 in. water[a]	5.0

[a]An equivalent shield may be made with 2 in. of Benelex 70 or 6 in. of normal concrete.
[b]An equivalent shield may be made with 16 in. of normal concrete.

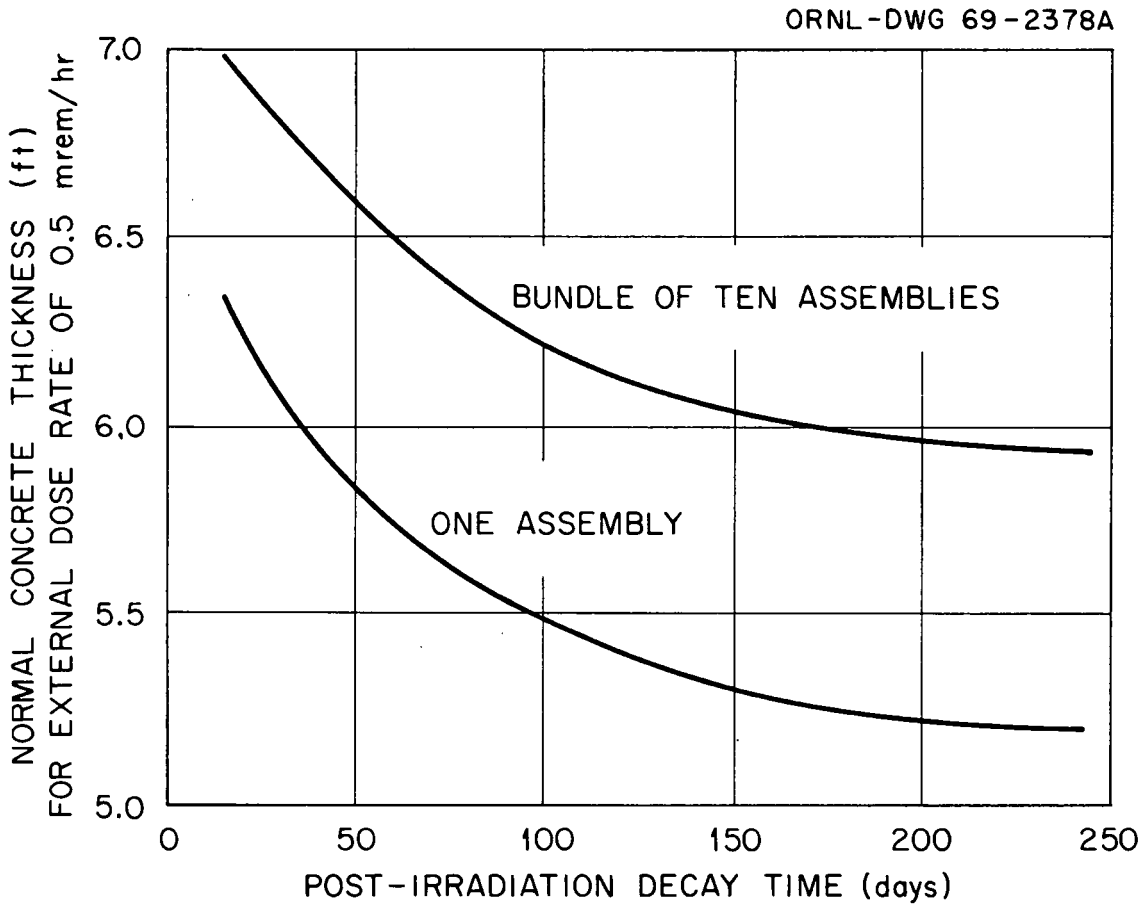

Fig. 4.19. Normal Concrete Shield Requirement (0.5 mrem/hr External Dose Rate) for LMFBR Core Fuel Assemblies as a Function of Decay Time. Each assembly has an active length of 4 ft, is located 3 ft from the inner concrete wall, and contains 50 kg of fuel irradiated to a burnup of 100,000 Mwd/metric ton at an average specific power of 125 Mw/metric ton.

Shield Requirements in Plutonium Product Handling. - A plant for processing spent LMFBR fuel will contain equipment for final purification of the plutonium product and for shipment of this product to a fuel preparation plant. In present practice, the plutonium product from spent fuel processing plants is shipped as nitrate solution. It is likely that future plants for processing LMFBR fuels will tend toward conversion of the plutonium product to an oxide since dry oxide can be shipped with greater safety and at lower cost than a liquid. It is also possible that the plutonium would be mixed with UO_2 and that the final product of the fuel processing plant would be a UO_2-PuO_2 powder suitable for shipment to a plant for fabrication of LWR or LMFBR fuel elements.

Shielding requirements for process equipment lines for final purification of plutonium, conversion to oxide, or preparation of UO_2-PuO_2 have been estimated by making use of data that had been obtained during the preparation of $^{233}UO_2$-ThO_2 fuels.[19] Table 4.21 presents estimates of the maximum capacities of lines for preparing two kinds of UO_2-PuO_2 fuels as a function of shielding thickness. In lightly shielded glove-box lines, it is estimated that LMFBR (~20% Pu) and LWR (~4% Pu) fuels can be prepared at maximum rates of 10 and 50 kg/day (3 shifts/day), respectively. The rates increase by about a factor of 4 if the operation is made semi-remote with the installation of 1-ft-thick concrete shielding (although it is probably unreasonable to fabricate LWR fuel at a rate of 200 kg/day by this technique). It is envisioned that plants for processing these fuels at rates greater than about 100 kg/day would employ highly automated and remotely operated equipment and that there would be no appreciable penalty in using concrete shields on the order of 2 ft thick.

4.11.5 Required Developmental Work

Previous sections have summarized current knowledge concerning the radiation and shielding properties of LMFBR fuels that affect shipping and processing of the spent fuel and isolation and handling of the plutonium product. Significantly, additional data with increased accuracy will be required as LMFBR process development programs begin to focus on the most

Table 4.21. Estimated Maximum Capacities of Lines for Preparation of UO_2-PuO_2 as a Function of Reactor Type and Shield Thickness

Operator dose rate = 40 mrem/week.

Type of Operation	LMFBR Core LWR Recycle Pu		LWR-Pu Core LWR Recycle Pu	
	Concrete Shield Thickness (in.)	Capacity kg (UO_2-PuO_2)/day	Concrete Shield Thickness (in.)	Capacity kg (UO_2-PuO_2)/day
Gloved enclosures[a]	0[b]	10	0[b]	50
Semiremote[a]	12	40	12	(200)
Remote[c]	15	100	9	100
	21	500	15	500
	23	1000	17	1000

[a]Assumes process equipment installed 1 ft from inner wall of enclosure and dose results by analogy with ^{233}U-ThO_2 experience.

[b]Assumes 1/8 in. of lead for shielding soft gamma radiation.

[c]Assumes that process vessels (which contain no more than 20% of daily throughput) are installed 2 ft from the inner surface of the shield and that the dose rate at the operating surface is 1.0 mrem/hr.

attractive concepts, and as the concepts proceed toward the design of pilot and commercial fuel processing plants.

Nuclear Data. - Experimental and theoretical programs are required to develop data on cross sections, fission yields, thermal power, and radiation spectra of isotopes (such as those listed in Table 4.15) that might otherwise be overlooked in developing data for the design of LMFBR's. Continuing theoretical studies are required to correlate and fully utilize the integral and differential neutron cross-section data that will be generated in planned experimental and prototype irradiation programs. Improved methods should be developed for averaging these cross sections over LMFBR spectra and calculating concentrations of the most important nuclides in spent fuel. Improved thermal power and gamma yield data, including the effect of bremsstrahlung radiation, should be developed for many fission products to provide more realistic estimates of cooling and shielding requirements and radiation damage. Experimental and theoretical work is required to develop fission product yield data for the plutonium isotopes and ^{238}U in typical LMFBR spectra. Present fast-fission-yield data, predominantly for ^{238}U and ^{239}Pu, generally apply to discrete fission by neutrons with energy above 1 Mev.

Reactor Irradiation Studies. - Detailed nuclear calculations should be performed for the LMFBR designs of principal interest, and the resulting spectral data should be used to re-estimate concentrations of all of the important nuclides in spent fuel. Codes for calculating concentrations of radioactive materials should be developed in conjunction with the development of techniques for the nuclear design of LMFBR's. The nuclear data should be continually upgraded as results of critical experiments, prototype irradiations, and differential cross-section measurements become available.

Radiation Dose Rates and Shielding Requirements. - Progressively improved assessment of radiation dose rates and shielding requirements will be required during conceptual, preliminary, and detailed design of LMFBR fuel processing plants. Shielding requirements, as a function of capacity and type of operation, are particularly needed in the area of plutonium product handling and fuel preparation.

4.12 Criticality Studies - Task 12

Criticality considerations have high priority in the design and operation of facilities which process fissionable isotopes so that inadvertent criticality does not occur. This may be accomplished by maintaining limits on geometry, mass, fissile concentration, density, moderation, reflection, fixed or soluble neutron absorber concentration, or a combination of these. It is the goal of criticality studies to find the limits or combinations of limits for each step in a process which are safe, practical, and economical.

4.12.1 Basic Criticality Data

Homogeneous Systems. - Basic criticality data up to early 1964 has been summarized by Paxton *et al.*[1] for the fissile isotopes including ^{239}Pu. In Figs. 27 to 30 of ref. 1, the sphere critical mass, the sphere volume, the diameter of infinite cylinders, and the thickness of infinite slabs of homogeneous water-plutonium systems are given as a function of plutonium density or H/Pu atomic ratio. These curves are derived from results of early experiments at Hanford, Rocky Flats, Saclay, and Los Alamos, and corrections were applied so that the neutron absorption effects of nitrate and ^{240}Pu were not included. A set of experiments for 5 wt % Pu-Al rod lattices was also included in tabular form. Since then, efforts have been made toward generating basic criticality data, using nitrate solutions (having varying ^{240}Pu content) in spherical, cylindrical, and parallelopiped geometries. Calculations for these geometries, as well as the corrected experimental data, have been summarized by Hansen.[2] These data include PuO_2-polyethylene compacts at H/Pu atomic ratios of 5 to 15, which provide more confidence in the calculations for the H/Pu range below 50. Complete tabulation of experiments which have been analyzed,[2,3] performed in cylinders,[4,5] spheres,[6] and parallelopipeds,[2] are presented elsewhere. In addition, there exist critical experiments with one and two annular cylinders[7] of nitrate solution which may have application to special problems although the experiments were done with limited volumes. A precise measurement of the critical mass for a water-reflected

plutonium sphere has been completed.[8] Calculations of k_{eff} for this assembly, using transport theory, DTF, in the S_8 approximation and with Hansen-Roach cross sections, gave a value of 1.0037.

Richey[3] has calculated bare and water-reflected critical sphere radii for $Pu(NO_3)_4$ solutions 0, 3, or 6 M in nitric acid and containing 0 or 5 wt % ^{240}Pu. Hansen[2] has calculated water-reflected critical radii for spheres of Pu-H_2O mixtures for 0, 5, 10, 15, and 20 wt % Pu. Hansen and Clayton[9] have calculated the criticality conditions for various plutonium compounds for H/Pu ratios of 0 to 20.

Fixed Neutron Absorber. - Criticality experiments with borosilicate-glass raschig rings have not been done to determine the plutonium concentrations for which $k_\infty < 1$ for variations in boron content in the glass, for various volume fractions of glass, for various ring sizes, for various nitrate concentrations, or for various ^{240}Pu contents. A single neutron source multiplication experiment[10] has been performed in a 30-in.-diam cylinder. The raschig rings contained 19 wt % B_2O_3 (6 wt % boron) and filled 25% of the volume. The extrapolation of the inverse of the neutron source multiplication curve indicated that the test vessel would not be critical at any height for a nitrate solution that contains 350 g of Pu per liter and about 5.5 wt % of ^{240}Pu, and is 9 M in excess nitric acid.

Experiments using small cylindrical vessels, containing either randomly arranged raschig rings or regularly spaced Pyrex glass tubes, are in progress in France.[11]

Soluble Neutron Absorbers. - Criticality experiments with soluble neutron absorbers such as boron, cadmium, or rare earths in homogeneous plutonium solutions have not been performed.

Pu-Al Lattice Experiments. - Lattice experiments with three sets of Pu-Al alloy rods in light water have been reported by Hill and Parker.[12] The rods have 1.8, 2.0, and 5.0 wt % Pu in the alloy. Most of the data have been analyzed by Worden, Liikala, and Reardon.[13] The authors state:

> "Three areas of potential difficulty in the analysis of plutonium systems were studied; (1) the basic cross section data, (2) calculation of the space-energy distribution of thermalized neutrons in

a lattice cell, and (3) methods of calculating the neutron absorption rate in the 1.056 eV resonance of ^{240}Pu."

The calculated k_{eff}'s were within 0.983 to 1.026 for the Sher-Felberbaum[14] thermal cross sections and within 1.005 to 1.054 for the Leonard[15] values for the experiments for which k = 1.000.

Rods containing 5 wt % plutonium are currently being studied in lattice experiments with boron dissolved in the moderator.[16]

Plutonium lattice experiments are important because they permit the self-shielding effects for neutron absorption in the 1.0-ev resonance of ^{240}Pu to be understood. There may be a rod size for which the critical mass is smaller than for the homogeneous system, similar to effects found in low-enrichment uranium systems.

PuO_2-UO_2 Lattice Experiments. - Mixed plutonium-uranium oxide rod lattices have been studied at Battelle Northwest (formerly Hanford) Laboratories and at the Westinghouse Reactor Evaluation Center (WREC). Although these lattices contain much less plutonium and more uranium than expected in the LMFBR fuels, the methods for calculating these experiments will be useful for LMFBR compositions in rod lattices. The early Hanford experiments[12,17] with 1.5 wt % PuO_2 in depleted UO_2, and the WREC experiments[18] with 6.6 wt % PuO_2 in natural UO_2 have been calculated by Bindler and Poncelet.[19] The results are in good agreement; in general, k values are greater than unity. If criticality calculations are made for k = 1 for lattice arrangements, conservative sizes will result. Schmid[17] reports calculated masses 2 to 5% higher than obtained experimentally.

Orr et al.[20] reported calculated k_{eff} values for the WREC lattice experiments within -0.01 to + 0.02 of the experimental value of 1.000.

The initial loading for the unrodded core in EBWR program for the utilization of plutonium was calculated[21] to have a k_{eff} of 1.007 for 22 fuel assemblies, each of which contained 1.5 wt % PuO_2-UO_2.

Lattice experiments[22,23] with 2.0 wt % PuO_2 (plutonium isotopic composition: 7.65, 16.64, and 23.50 wt % ^{240}Pu) in natural UO_2 have been completed, and calculations for these underestimate the k_{eff} values

by 1 to 2%. The Sher-Felberbaum[14] thermal cross sections were used; however, substituting recent cross-section[24] values increased k_{eff} approximately 1%.

Criticality Calculations for LWR and LMFBR Fuels. - The problem of selecting criticality controls that will not unduly restrict fuel cycle processing options is magnified by the paucity of critical experimental data for systems that contain high concentrations of ^{240}Pu, ^{238}U, and other neutron absorbers. Some calculated criticality data[25] that will be useful for the development of preliminary equipment designs have been generated. The criticality calculations were performed, using the transport theory code, ANISN, in the S_4 approximation with 16-group Hansen-Roach cross sections. Cross-section sets for ^{241}Pu and ^{242}Pu not found in the Hansen-Roach library were taken from the GAM-II library. The surveys were performed over the range of moderation to show critical sizes and masses primarily for the discharge fuel compositions anticipated in LMFBR's and LWR's. Cadmium nitrate or boric acid at several different concentrations was included in some of the computations to show the increased critical sizes resulting from the presence of neutron absorbers, especially in the high moderation range. Additional calculations for various fuel oxide mixtures were made to provide guidance in other fuel element manufacturing or processing steps.

Selected plutonium critical experiments were also calculated.[25] They yielded k_{eff}'s less than 2% high except for the PuO_2-polystyrene compacts, where the k_{eff}'s were 3 to 5% larger than unity. For the latter, no correction was made for the effects of particle size in the PuO_2, and the agreement is considered to be satisfactory. The calculated[25] k_{eff} was 1.0044, using ANISN transport code for the water-reflected plutonium sphere.[8]

Calculations were made over the range of H/Pu atomic ratios for oxide-water mixtures of ^{239}Pu, LWR Pu, LMFBR Pu, LMFBR (Pu+U) fuels [Pu/(Pu+U) = 0.247 and 0.243]; the LMFBR fuels had various concentrations of boron and cadmium. Calculations were also made for plutonium fuels having only two isotopes; the ratio of these isotopes was varied. Typical

parameters, including the effects of ^{240}Pu, ^{238}U, boron, and cadmium, are illustrated by Figs. 4.20-4.24.

The 05R Monte Carlo code[26] was used to calculate k_∞ for five plutonium nitrate solutions; two sizes of borosilicate-glass tubes containing 4 wt % boron were considered. The solutions contained excess nitric acid (the nitric acid concentration varied from 0.74 M to 2.8 M) and 132 to 431 g of ^{239}Pu per liter. The 1.50-in.-OD x 1.25-in.-ID tubes at 24 vol % glass corresponded to a random arrangement of similar sized Raschig rings. The 0.625-in.-OD x 0.43-in.-ID tubes at 30 vol % glass corresponded to another Raschig ring size. The calculations show that k_∞ is less than 1 for solution--24% glass systems containing less than 220 g of plutonium per liter and for a solution--30% glass system containing less than 450 g of plutonium per liter. With 5 wt % ^{240}Pu, the critical concentrations are calculated to be greater than 450 g of plutonium per liter. Calculations also showed that the random arrangement of Raschig rings would be less reactive than the glass tube arrangement. These calculations are not inconsistent with the measurement by Schuske.[10]

4.12.2 Engineering Applications of Criticality Data

Spent Fuel Shipping. - The minimum critical mass of plutonium in a compact array of typical LMFBR core fuel assemblies flooded with sodium and reflected by steel or lead is not known accurately at present, but may be estimated to be in the range of several hundred kilograms.[27] Even larger masses would result if the elements were spaced slightly farther apart than in the reactor or interspersed with radial blanket elements. These estimates imply that, from a criticality point of view, a large fraction of an entire LMFBR core could be shipped at one time in a single cask if sufficient features are provided to assure that the sodium is not displaced by a more efficient moderator such as water.

It is anticipated that reactor critical experiments will provide sufficient information to determine the necessary criticality parameters of LMFBR casks that are filled with sodium or gas. Critical experiments, perhaps using boron -- stainless steel plates between assemblies or mockups

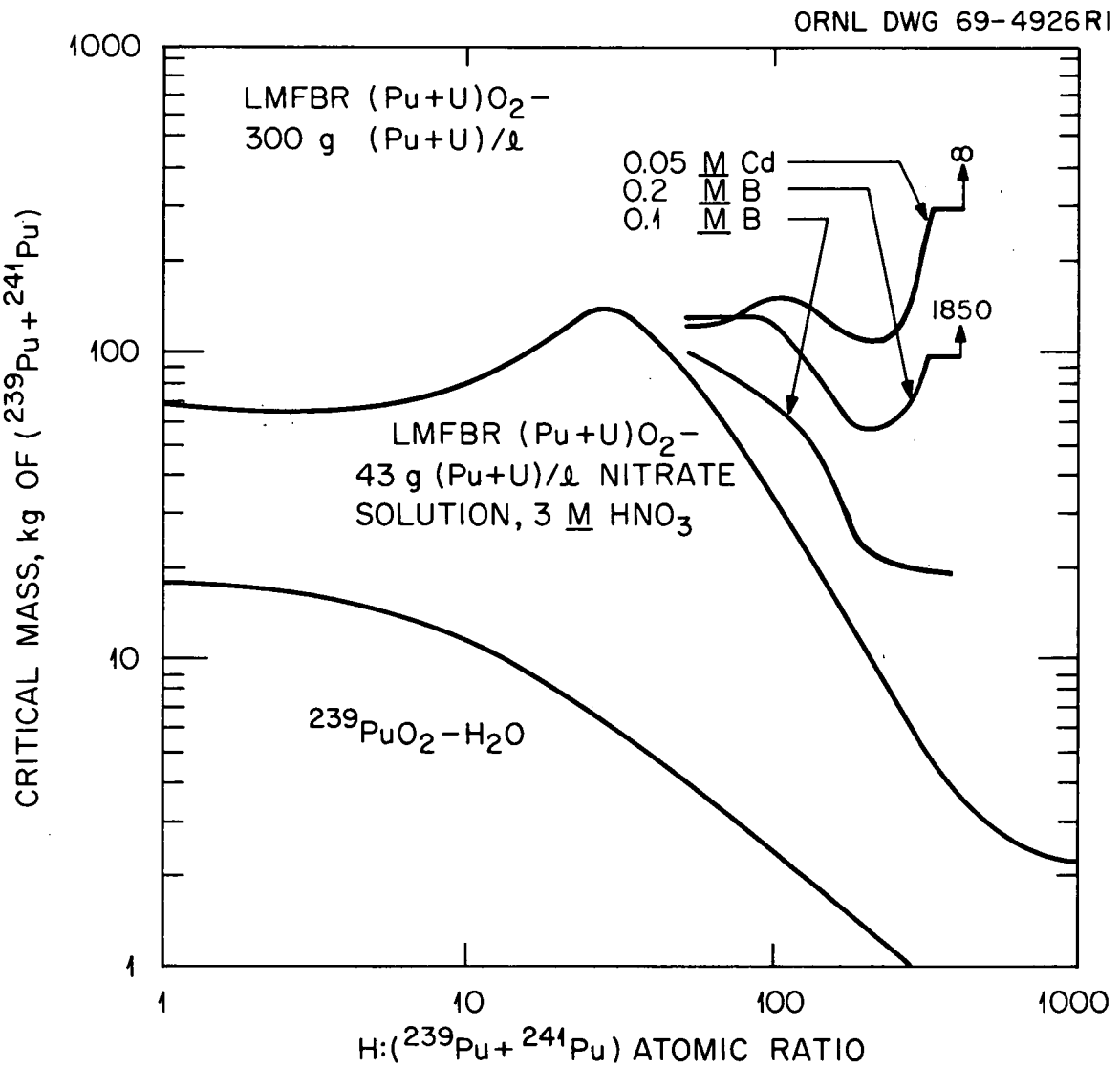

Fig. 4.20. Critical Masses of Spheres for Various Plutonium Fuels Reflected by 3-cm-thick Stainless Steel.

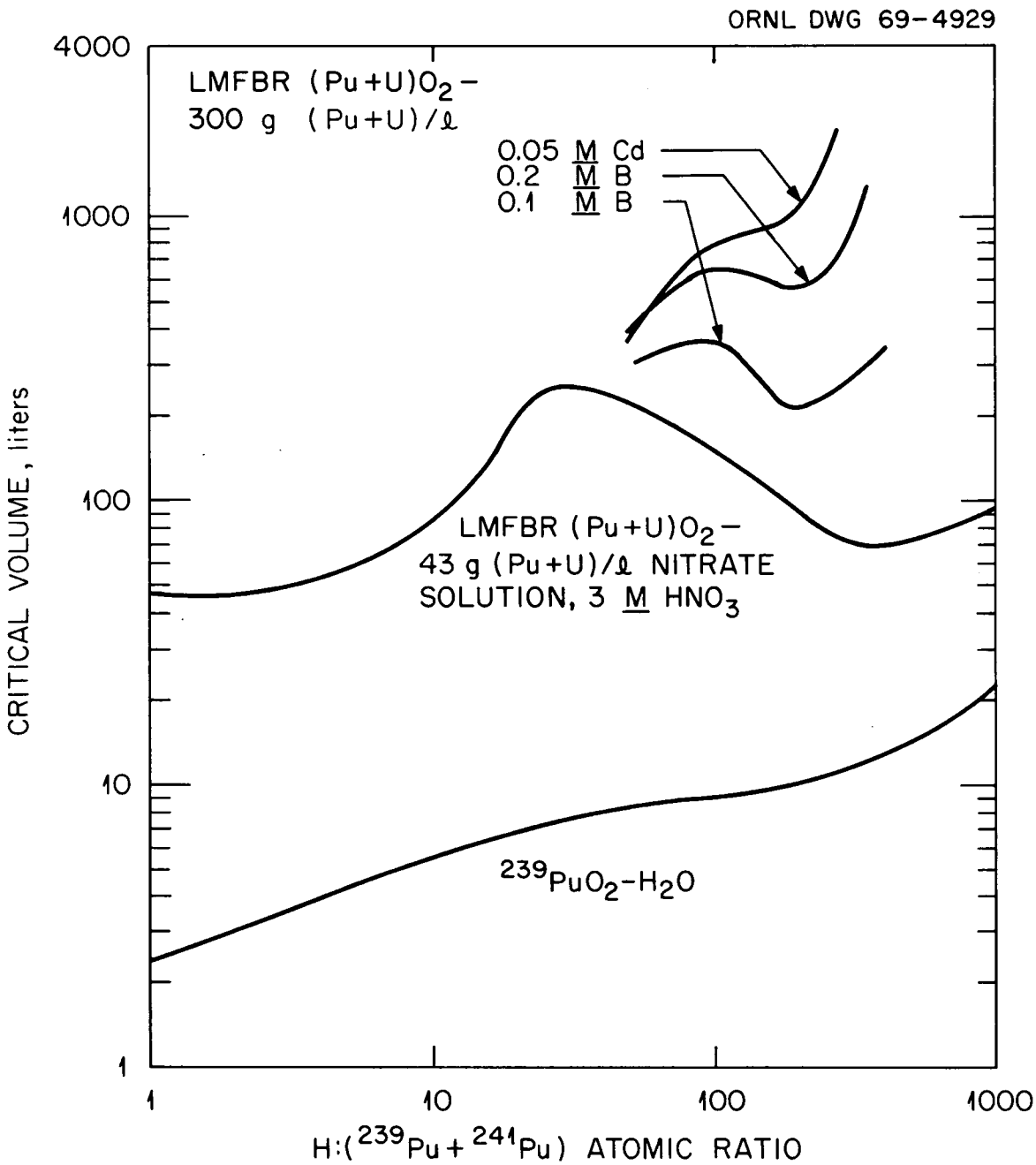

Fig. 4.21. Critical Volumes for Spheres of Various Plutonium Fuels Reflected by 3-cm-thick Stainless Steel.

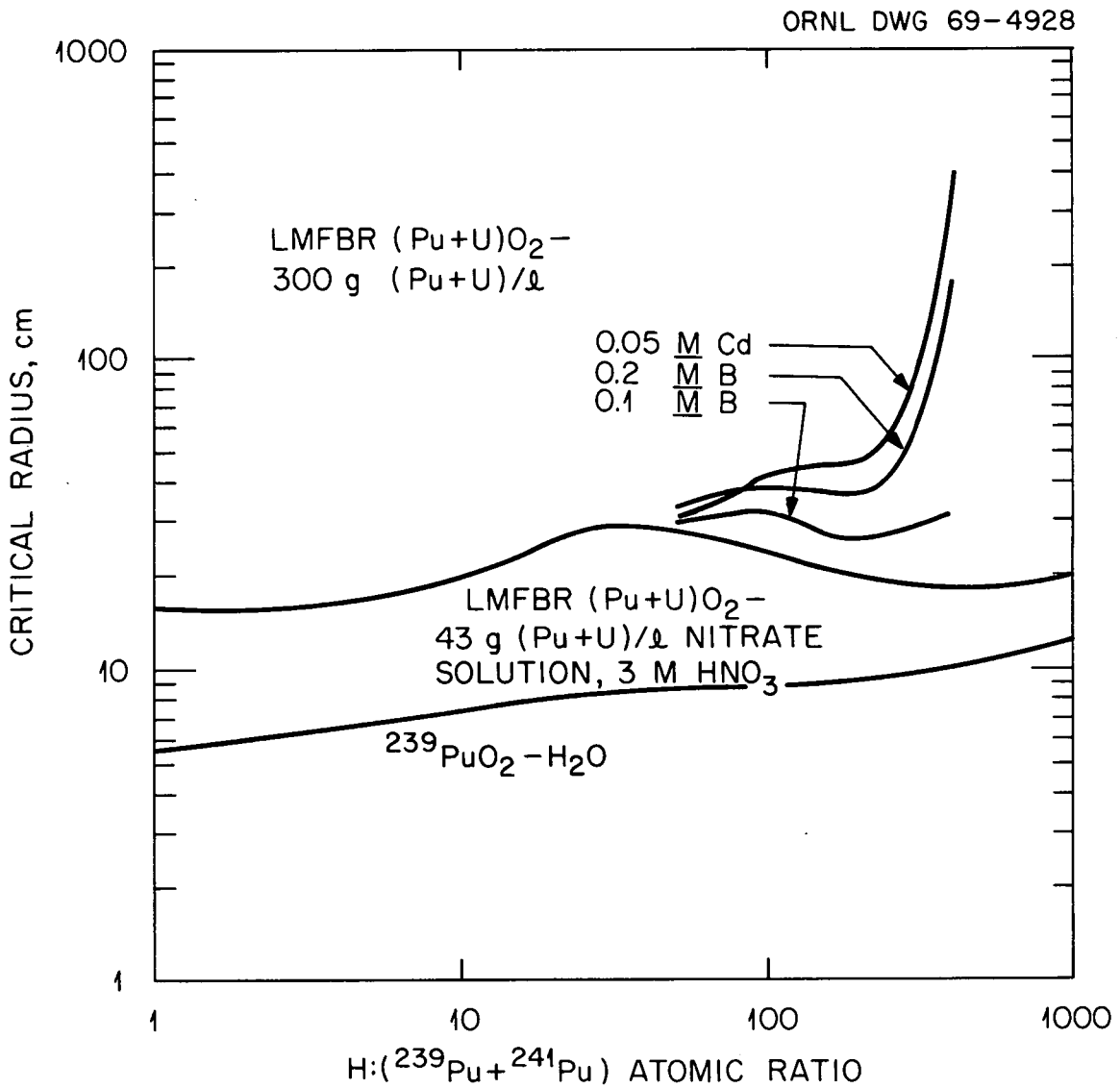

Fig. 4.22. Critical Radii for Infinite Cylinders of Various Plutonium Fuels Reflected by 3-cm-thick Stainless Steel.

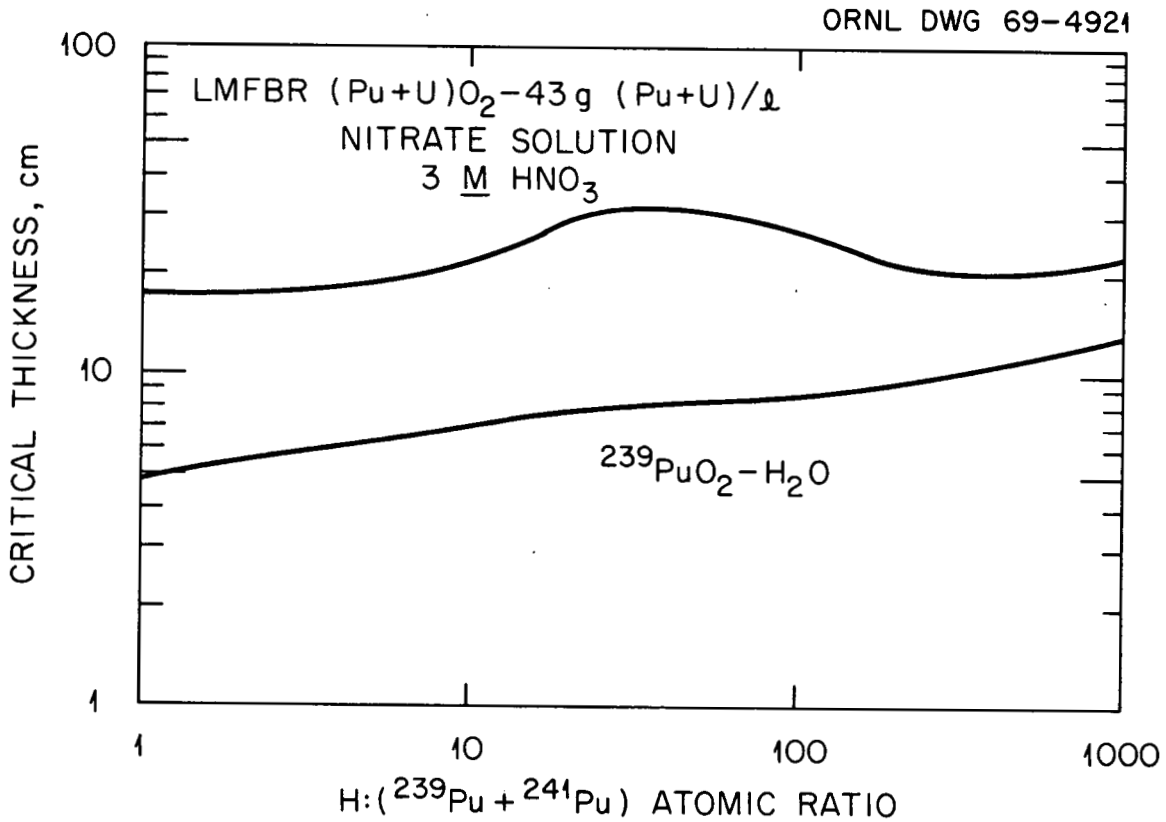

Fig. 4.23. Critical Thicknesses of Infinite Slabs for Various Plutonium Fuels Reflected by 3-cm-thick Stainless Steel.

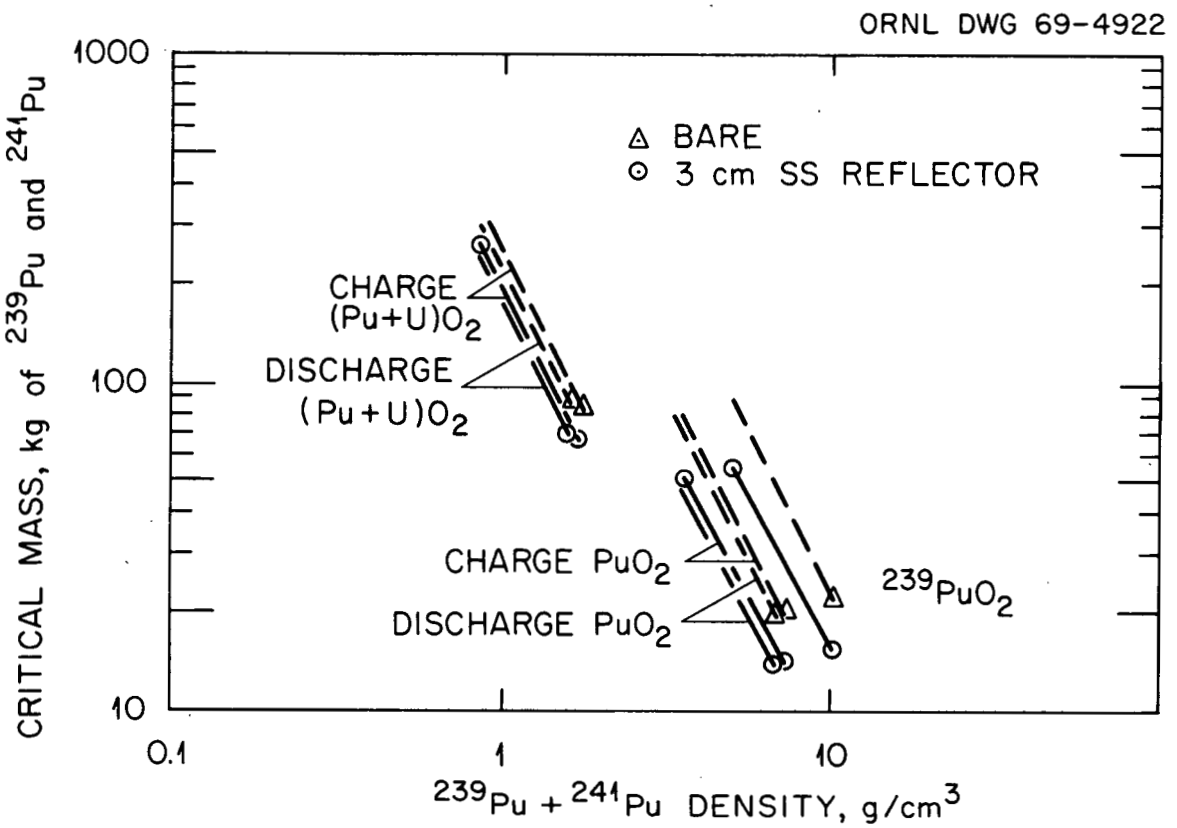

Fig. 4.24. Critical Masses for Spheres of $^{239}PuO_2$, LMFBR Charge PuO_2, and LMFBR Discharge $(Pu+U)O_2$ at Various Densities.

of assemblies, will be necessary if subcriticality must be assured in the event of water moderation.

Spent Fuel Storage, Handling, and Chopping. - Minimum critical masses of plutonium in dry, chopped fuel or arrays of elements are in the hundreds of kilograms and increase as the inverse 1.0 to 2.0 power of the apparent density of the fuel (Fig. 4.24). No significant criticality limitations are expected in dry or liquid-metal-cooled storage, handling, and chopping of LMFBR elements. It may become necessary to limit the number of elements during the steaming of equipment for removal of sodium.

Fuel Element Dissolvers. - The calculations summarized in Figs. 4.20-4.24 were performed primarily to aid in the selection of conceptual designs for LMFBR fuel element dissolvers. Critical masses, volumes, and dimensions were calculated over virtually the entire range of possible hydrogen moderation in an attempt to estimate minimum critical parameters for all possible suspensions of undissolved fuel solids in solutions of partially dissolved fuel. A summary or these estimated critical parameters is presented in Table 4.22. It has been assumed that the settling of fuel particles in dissolvents will result in solid volume fractions no greater than 0.6.

The first five rows of Table 4.22 illustrate that suspensions of pure PuO_2 in water or dilute dissolvent can result in a minimum critical fissile mass of 1 kg or less, minimum critical volumes of only several liters, cylinder diameters of less than 5 in., and slab thicknesses of less than 3 in. Such limitations on these parameters are potentially required under conditions in which the UO_2 might be leached preferentially from the UO_2-PuO_2 (e.g., in the dissolution of unirradiated, mechanically blended pellets) or if plutonium polymer formation is possible. Line 6 of the table shows, however, that the effect of making the dissolvent 0.4 M in cadmium is to materially increase these parameters.

The remaining 7 rows of Table 4.22 show the appreciably larger values of the minimum critical parameters that will result with the use of cadmium as a soluble poison if it can be assumed that the UO_2-PuO_2 fuel particles will not materially change in composition during dissolution. The effect

Table 4.22. Calculated Critical Parameters for 60 vol % Fuel Oxide Solids in Various Solutions for Various Reflector Conditions

Fuel Solid	Solution[a]	$H/(^{239}Pu + ^{241}Pu)$ for 60% Solids	Reflector		Critical Sphere[b]		Infinite Cylinder Diameter (in.)	Infinite Slab Thickness (in.)
			Material	Thickness (cm)	Mass (kg of $^{239}Pu + ^{241}Pu$)	Volume (liters)		
$^{239}PuO_2$	Water	1.75	Bare		27.2 (0.86)	4.6	6.1	3.3
$^{239}PuO_2$	Water	1.75	SS	3	17.5 (0.62)	2.9	4.6	2.2
$^{239}PuO_2$	Water	1.75	Water	15	11.2 (0.48)	1.8	3.5	0.9
LWR-PuO_2	Water	2.41	SS	3	17.0 (1.0)	3.8		
LMFBR-PuO_2	43 g (Pu+U)/liter	2.39	SS	3	18.5 (~ 1)	4.6	5.5	2.6
LMFBR-PuO_2	300 g (Pu+U)/liter, 0.4 M Cd	2.05	SS	3	20.0	5.0		2.7
LMFBR-$(Pu+U)O_2$	43 g (Pu+U)/liter	10.2	SS	3	84. (2.3)	86. (65.)	15.7	8.6
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 0.2 M Cd	8.74	SS	3	90.	92.	15.9	9.0
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 0.4 M Cd	8.55	SS	3	96.	98.	16.1	9.2
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 1.0 M Cd	7.89	SS	3	110.	112.	17.3	9.8
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 0.2 M Cd	8.74	Soln.	15[c]	72.	74.		
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 0.4 M Cd	8.55	Soln.	15[c]	86.	88.		
LMFBR-$(Pu+U)O_2$	300 g (Pu+U)/liter, 1.0 M Cd	7.89	Soln.	15[c]	106.	108.		

[a]All solutions (except water) were 3 M in HNO_3.

[b]Values in parentheses are minimum values at higher H/Pu ratios; other values are minimal at 60 vol %.

[c]An infinite reflector of solution was simulated by returning all leakage neutrons at 15 cm. Plutonium in reflector not included in the mass value.

of the presence of the $^{238}UO_2$ is to increase the critical parameters in the low moderation range in which the cadmium absorption is not effective. The presence of stainless steel hulls mixed with the oxide fuel would enhance this effect, but the hulls were omitted in this analysis for conservatism. The effect of the cadmium in solution is to provide that no more restrictive limits will apply in the intermediate range of moderation. In principle, boron in solution would serve this purpose to a lesser extent. Natural boron, near its solubility limit of approximately 0.2 M, has neutron absorption properties comparable to cadmium at about 0.05 M.[28] The dispersions of UO_2-PuO_2 in dissolvents containing cadmium have fissile masses greater than 70 kg; minimum critical cylinder diameters and slab thicknesses are in the range of 15 and 9 in., respectively.

Present data indicate that selective leaching of UO_2 from irradiated UO_2-PuO_2 is probably negligible, but this must be substantiated for all appropriate conditions to provide confidence in the application of these more favorable values of the minimum critical parameters. A variety of selected critical experiments should be performed to validate the methods for computing critical parameters of dispersions of mixed UO_2-PuO_2 solids in solutions containing cadmium and boron. In addition, critical experiments should be performed to validate specific equipment designs as LMFBR dissolution hardware development approaches plant scale.

Pulsed Extraction Columns. - The Nuclear Fuel Services fuel processing plant employs a 10-in.-diam extraction column to process (U + Pu) fuel, using the Purex process, at the rate of about one ton/day. The expanded top section of the column contains plates of 1% boron--stainless steel to make the column safe at the anticipated maximum plutonium concentration. The rationale for the use of a 10-in.-diam column generally assumes that administrative control will be exercised to prevent the effect of reflux from concentrating the plutonium in the column to greater than about 35 g of plutonium per liter. A more conservative assumption, i.e. that reflux is not detected before the solvent becomes saturated with plutonium (at about 120 g of plutonium per liter), would require that column diameters be limited to about 8 in.

Boron--stainless steel is not considered as a suitable material of construction for pulse plates (which are subject to vibration and difficult to inspect periodically) because this material has poor resistance to corrosion and requires mechanical (rather than welded) fasteners due to its brittleness. Developmental work for reactor control plates has revealed that stainless steels containing a dispersion of gadolinium oxide in the concentration range of interest do not suffer any appreciable change in their mechanical and chemical properties.

The calculations presented in Fig. 4.25 were performed to estimate the critical diameters of pulsed columns that would use pulse plates made of gadolinium--stainless steel.[29] A plate thickness of 1/8 in. was assumed for most of these calculations in an attempt to estimate the effect of the normally 3/16-in.-thick plates which have holes in 23% of the sectional area. From these data it is estimated that the effect of incorporating pulse plates containing about 0.3 wt % gadolinium at a 1.5-in. plate spacing would be to provide essentially unlimited column diameters for the customary assumption of a maximum plutonium concentration of about 35 g/liter [$H/(^{239}Pu + ^{241}Pu) = 1000$] or to approximately 9.5 in. for the more conservative assumption of a maximum plutonium concentration of ~120 g/liter [$H/(^{239}Pu + ^{241}Pu) = 220$]. At a 1-in. spacing and 0.4 wt % gadolinium, the critical column diameters are about 16 in. at 80 g of plutonium per liter and about 11.8 in. at 120 g of plutonium per liter.

Such large column sizes can contribute to the economy of plants for processing power reactor fuels (LWR as well as LMFBR) at rates of 5 tons/day or greater. Confirmation of the estimates presented above would require critical experiments to validate the poisoning effect of gadolinium and to determine the geometrical effects of the holes in the pulse experiments to validate the corrosion resistance of the gadolinium stainless steel, and studies to generate designs that will ensure priate plate spacing and integrity during the intended service li

Plutonium Storage. - The use of large tanks packed with boro glass Raschig rings is currently the preferred technique for inte storage of purified plutonium solutions in processing plants. Ev

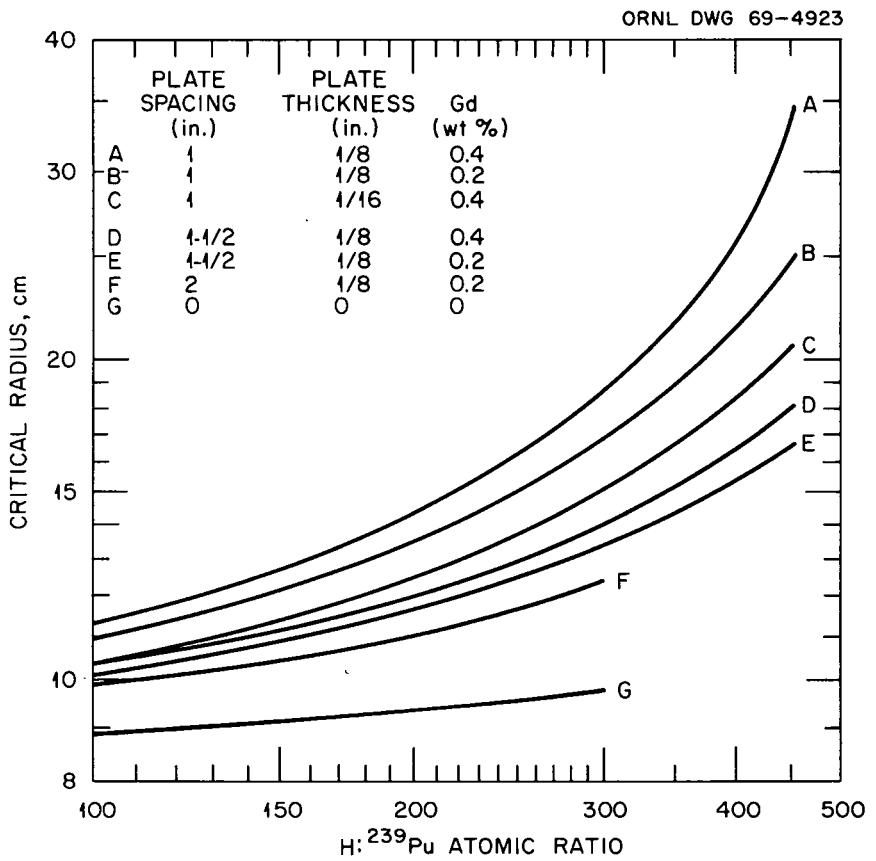

Fig. 4.25. Critical Radii for Pulsed Extraction Columns That Have Gadolinium--Stainless Steel Plates and Are Reflected by 3-cm-thick Stainless Steel.

good that solutions of the normal storage concentration, about 200 g of plutonium per liter, will remain subcritical and that the product would not become contaminated by leaching of boron from the rings. It has not yet been determined that borosilicate-glass raschig rings will have a suitable service life in solutions of high-exposure plutonium with associated high alpha and beta radiation levels.

Plutonium Product Shipment. - The shipment of plutonium solution in 10-liter polypropylene bottles contained in bird cages is currently considered safe; however, it is relatively expensive. Larger containers incorporating either high surface-to-volume geometry or fixed neutron sbsorbers could improve the economics of the shipment of solutions by decreasing handling and bulk shipping charges. Still larger and more economical shipments in a single container are possible if the plutonium is in the form of a dry oxide.

4.13 Design and Engineering Studies - Task 13

Because of the differences in fuels used in LMFBR facilities as opposed to LWR systems, the design features and design parameters required in a processing facility to recover fissile and fertile material from LMFBR fuels may well be considerably different from the design features found in a plant intended to recover these materials from spent LWR fuels. Safety of operation in an LMFBR fuel processing plant, as in an LWR processing plant, is a matter of prime concern. Because of the probable presence of sodium, the expected presence of relatively large quantities of radioactive fission product gases (including radioiodine), and the increased potential for attainment of nuclear criticality, all the features of plant design relating to safety and containment of fission products should be reviewed and studied in order to make an assessment of the best way to attain the desired level of safety at reasonable cost.

The duration of the decay period prior to recovery of the valuable components from spent fuel will have a more pronounced effect on the overall fuel cycle cost for the LMFBR than for the LWR. The present practice in recovering valuable materials from LWR fuel is to allow the fuel to

decay for a period of at least 150 days prior to processing. Unquestionably, decreasing this period of decay will increase the difficulty of recovering the valuable materials from LMFBR fuel. We do not know at this time the magnitude of the increase in capital investment and other charges for a given reduction in decay period. Because of the absence of positive information concerning the capability for recovery of fuel at short decay times, it will be necessary to delay an analysis of the economic decay period until a capability for short-decay processing is demonstrated. A need does exist, however, to determine the increase in investment that can be tolerated per unit reduction in decay time.

Spent LMFBR fuels will have plutonium contents that are approximately an order of magnitude greater than are found in LWR fuels. These increased plutonium contents may result in greater problems with plutonium polymerization and plutonium handling in general. We know of no thoroughgoing study dealing with problems involved in handling high concentrations of plutonium or the relatively large mass of plutonium to be processed daily.

The trend in LWR fuel recovery is to employ a central facility to process fuel from many reactors. This involves shipment of the spent fuel from the reactors to the central recovery facility and, subsequently, shipment of the decontaminated plutonium from the recovery facility to a fabrication facility and thence to the reactors. Because of the increased difficulty of shipping spent LMFBR fuel and because of the differences in the fuel itself, as well as the possibility of the advent of very large reactor stations at some time in the future (let us assume ca. 1990), the possibility exists that a plant located at the site of the reactor station and serving essentially only that reactor station might be economically justified. Under the conditions that are expected to exist with LWR's in the foreseeable future, it has not yet been found to be economically practical to situate a processing facility at a reactor station; however, the conditions that may prevail in the future and the differences in the nature of the fast reactor fuel may result in a reversal of this situation.

Nuclear safety will be considerably more difficult in an LMFBR fuel processing facility because of the increased quantity and concentration of

fissile material. We do not know of any in-depth studies that have been made relative to nuclear safety in recovery facilities for LMFBR-type fuels, either in an interim or a long-term recovery facility. The work done thus far has been of a scouting nature and has been directed toward the interim processing facility. It should be extended more fully in order to ensure satisfactory application in actual situations, including an interim processing facility. A great deal of additional work is indicated for the large-scale, long-term system in order to establish basic principles concerning the design and operation of process equipment.

Additional investigations are needed to determine what modifications in equipment and concept of operation would be required to permit recovery of LMFBR fuel in facilities that will exist in the 1980's. These studies should involve not only the necessary changes in equipment but also should deal with the method of operation; for instance, an assessment is needed to indicate whether the dissolved LMFBR fuel should be processed separately from LWR fuel, or whether the two might be blended together. No study of this problem has been made to date.

Certain other problems that are not entirely peculiar to LMFBR processing facilities deserve additional investigation, especially in view of the requirements of an LMFBR fuel recovery facility. These include accountability for source and fissionable material and a determination of the relative merit of continuous vs batch operation of certain items of process equipment (such as the dissolver). Also of interest are the effect of equipment size (as opposed to capacity) on the cost of the overall facility as well as on the cost of recovery of fuel, and of the effect of other general characteristics of equipment on processing cost. These items are matters of significant concern, but have not yet been adequately studied to permit a designer to make a clear choice of a particular operational system.

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5. STATEMENT OF PROBLEMS AND DEVELOPMENT PROGRAM TASKS

In this section the developmental work necessary to provide the information for the aqueous processing of LMFBR fuels is detailed, and a proposed schedule is presented. The new problems that will be encountered in the processing of LMFBR fuel can be divided into three categories:

A. Problems that are independent of decay time:
   1. Safety problems in handling fuels that may contain sodium.
   2. Mechanical problems owing to the fuel configuration, irradiation damage to metals, etc.
   3. Criticality and associated problems resulting from high plutonium concentrations in the fuel.

B. Special problems that are ameliorated or eliminated by delaying processing for possibly up to one year are:
   1. Retention of short-lived gaseous fission products (specifically ^{131}I and ^{133}Xe).
   2. Dissipation of decay heat during shipping and storage.
   3. Dissipation of decay heat during head-end operations.
   4. Radiation damage to solvents (although this is not likely to be a significant problem).

C. Problems that are made significantly more severe by increase in capacity are:
   1. Disposal of long-lived radioactive gases (^{85}Kr, $^{3}H_2$, $^{129}I_2$).
   2. Criticality.

In general, early emphasis should be placed on the first group, while the longer-term approach will be reserved for those problems that are occasioned by the economic incentive for short-decay processing and that are of such technical complexity as to require an extensive development period.

At this time, we cannot establish the optimum decay time or any other specific parameter of fuel condition, but we recognize the incentive for minimizing the decay period (one month of decay time is the economic equivalent of a 1% plutonium loss).

The breakdown of tasks outlines the proposed development in depth and covers a spectrum of parameters, such as fuel characteristics and cooling times, to provide the generalized basis for optimization of the processing cycle.

5.1 Shipping - Task 1

The objective of Task 1 is to ensure that an economic and safe method of shipment of LMFBR spent fuel will be available when needed for transport of fuel from the demonstration and early commercial LMFBR's and, later on, from the large number of commercial reactors. The problem involves finding improved methods of heat dissipation which are compatible with the available facilities and which are acceptable to operators and regulatory authorities. The heat generation rate and physical characteristics of the spent fuel cannot presently be stipulated; hence, it will be necessary to examine the problem over a sufficiently broad spectrum to allow future cask designers to optimize for whatever condition that might exist.

This work may involve analytical studies of the various facets of the problem: design, construction, and testing of components and assemblies; and preliminary design of prototype casks suitable for industrial needs.

The process options are depicted in Fig. 5.1. A logic diagram of the proposed work is given in Fig. 5.2, and a proposed schedule of activities is shown in Fig. 5.3.

Task 1.1 Evaluation of Heat Dissipation Methods

This subtask involves the comparison of the various possibilities for heat transfer and transport within a fuel shipping cask. The effects of the variation of fuel subassembly configuration and of the rate of decay heat generation may be determined. New calculational approaches and computer codes may be developed where there is a clear need, as determined by the absence of a suitable method for predicting performance with adequate accuracy and speed.

Task 1.2 Test of Heat Dissipation Methods

This subtask encompasses the work necessary to prove the performance of promising methods of heat dissipation and to provide basic information

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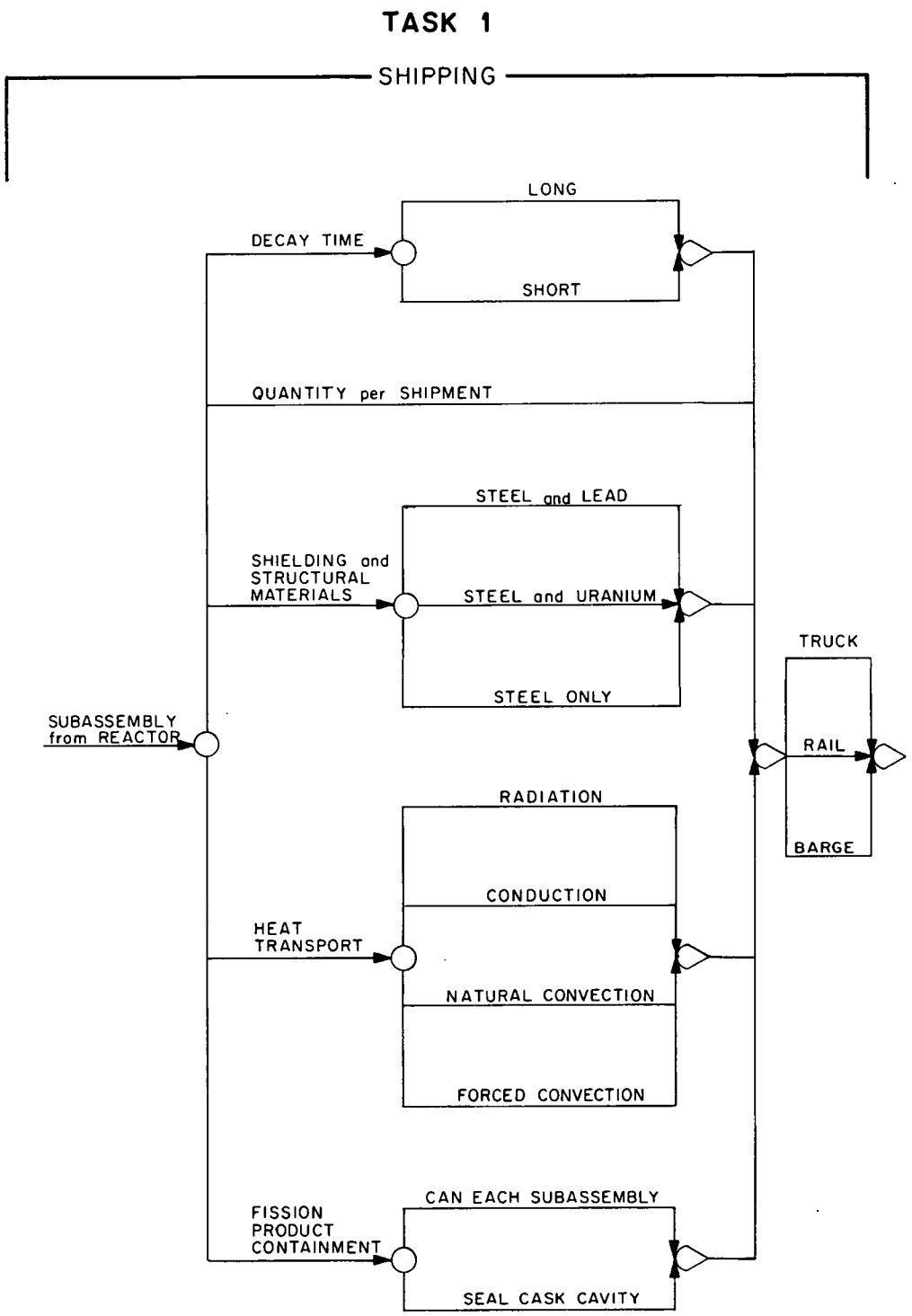

Fig. 5.1. Process Options for Task 1.

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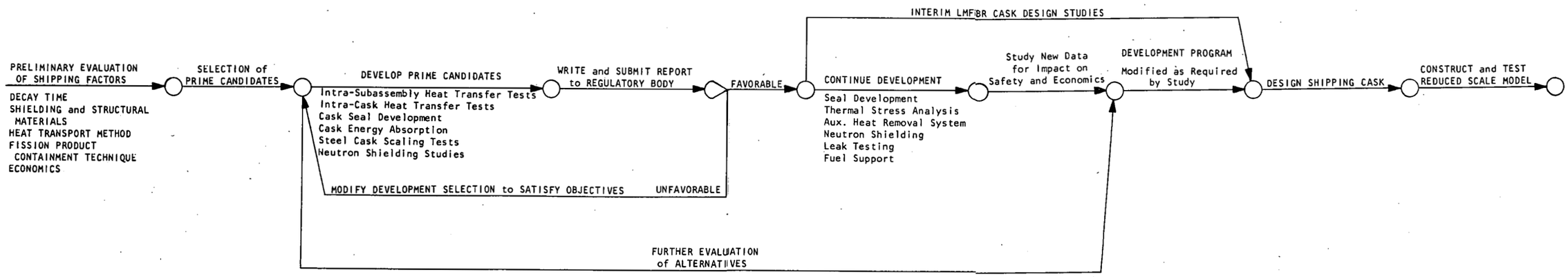


Fig. 5.2. Logic Diagram for Task 1.

ORNL DWG 70-4520

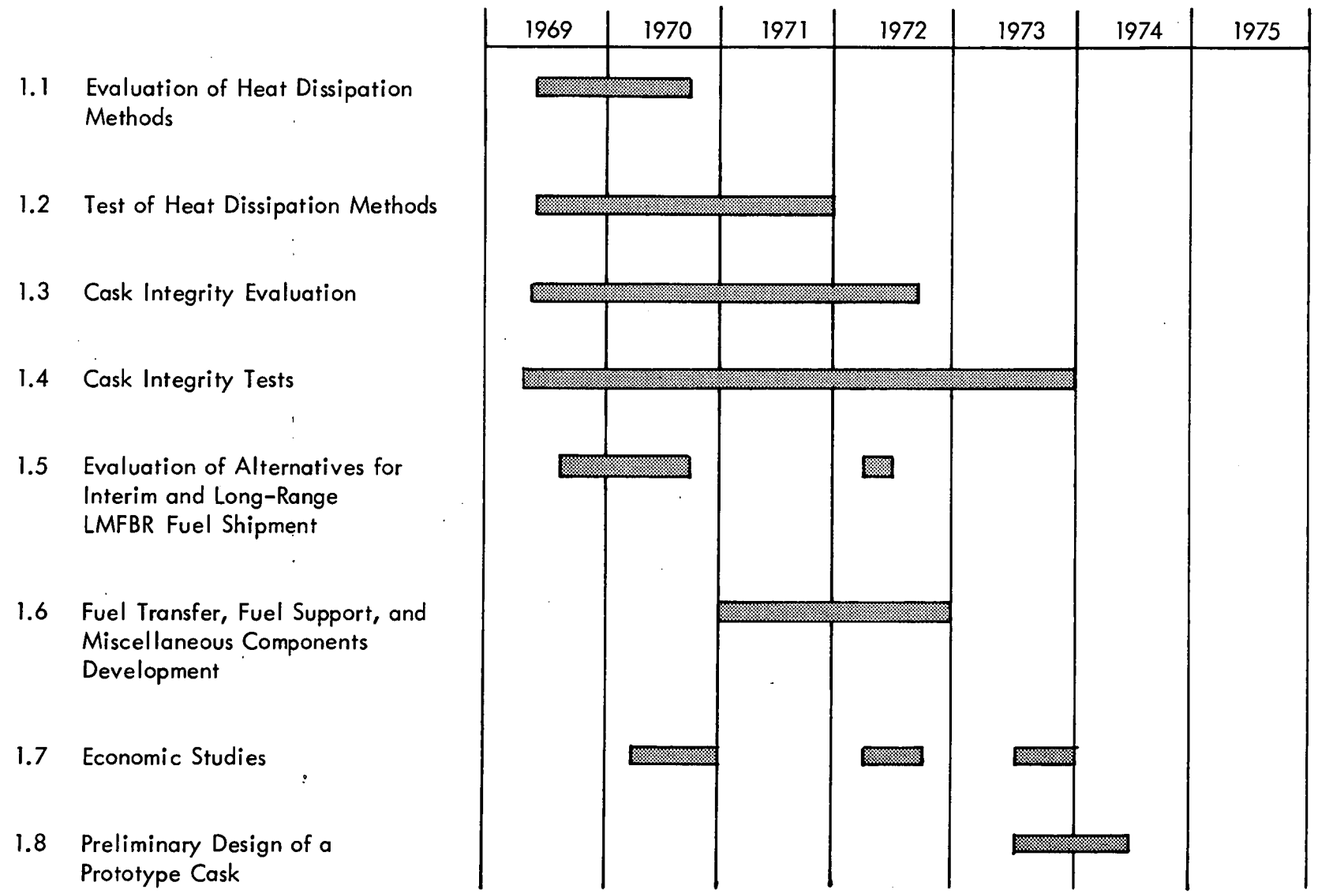


Fig. 5.3. Schedule of Activities for Task 1.

concerning promising methods of heat dissipation, whenever information of adequate reliability is not available.

Task 1.3 Cask Integrity Evaluation

This subtask covers analytical and literature survey work to understand the forces applied to a cask structure as a result of impact and of thermal effects. This work should include the selection of materials (to include composition, method of manufacture, and condition of heat treatment) and limitations on performance or applicability. A major portion of this effort should be concerned with seals for cask openings.

Task 1.4 Cask Integrity Tests

These tests should be directed toward determination and development of cask components and assemblies that can survive any credible accident. The minimum amount of protection necessary to provide satisfactory model performance should be determined for various components.

Task 1.5 Evaluation of Alternatives for Interim and Long-Range LMFBR Fuel Shipment

A study is required to determine the feasibility of using the same general system for the transport and receipt of fuels from demonstration reactors, early commercial reactors, and later, more numerous LMFBR's. This study should be made in conjunction with the study of receiving systems applicable to interim processing facilities (Task 2.2) and with general studies of interim processing. An evaluation should be made of various shipping systems that may become attractive as more is learned about heat dissipation, cask integrity, and fuel characteristics.

Task 1.6 Fuel Transfer, Fuel Support, and Miscellaneous Components Development

There are numerous problems dealing with the methods to be used for transfer of fuel into and out of a cask, support of the fuel in the cask, accommodation of coolant volume variation, and for testing to ensure that the fuel cavity is sealed. Work along these lines, of as general a nature as will be useful, should be carried out under this subtask.

Task 1.7 Economic Studies

The work performed under this subtask should provide the basis for the direction of that portion of the total task effort which is affected by economic considerations. Economic, or cost, studies should be made in conjunction with Tasks 1.5 and 1.8.

Task 1.8 Preliminary Design of a Prototype Cask

The goal of this task is to improve the economics and safety of LMFBR fuel shipment. The degree of success in this effort can be adjudged only by evaluation of a shipping system designed for a particular set of conditions and using all available information. This subtask includes the preparation of preliminary designs for shipping systems to be used during the interim processing period and for large-scale processing.

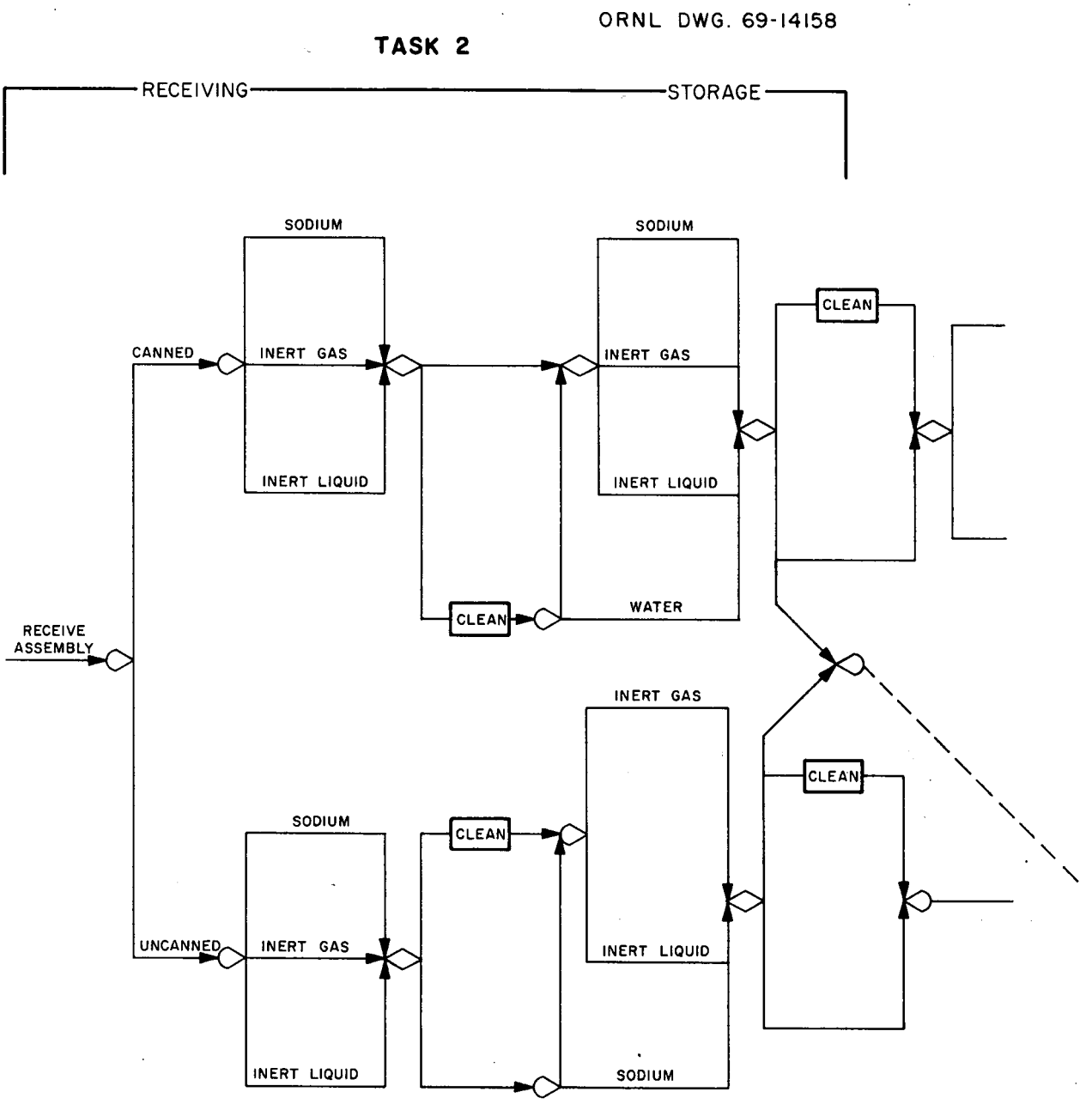
5.2 Receiving and Storage - Task 2

This subtask is concerned with the means for rapid, effective, economical, and safe operation of receiving and storage facilities for LMFBR fuels. The character of the work to be performed under this subtask will be determined largely by the outcome of investigations performed for the shipping and for the head-end processing tasks. Conversely, these other two tasks will be required to take into consideration the effect of variables in their areas on the task of receiving and storage. This work should take cognizance of related work on fuel handling and sodium removal to be performed under Elements 3 and 5, respectively, of the LMFBR Program Plan.

The process options are depicted in Fig. 5.4. A logic diagram of the proposed work is given in Fig. 5.5, and a proposed schedule of activities is shown in Fig. 5.6.

Task 2.1 Liaison with Elements 3 and 5

This subtask involves maintaining liaison with other organizations doing work on fuel handling and sodium removal. The objective is to make full use of all available information in this field.



NOTE: BROKEN FUEL ASSEMBLIES WILL BE CANNED

Fig. 5.4. Process Options for Task 2.

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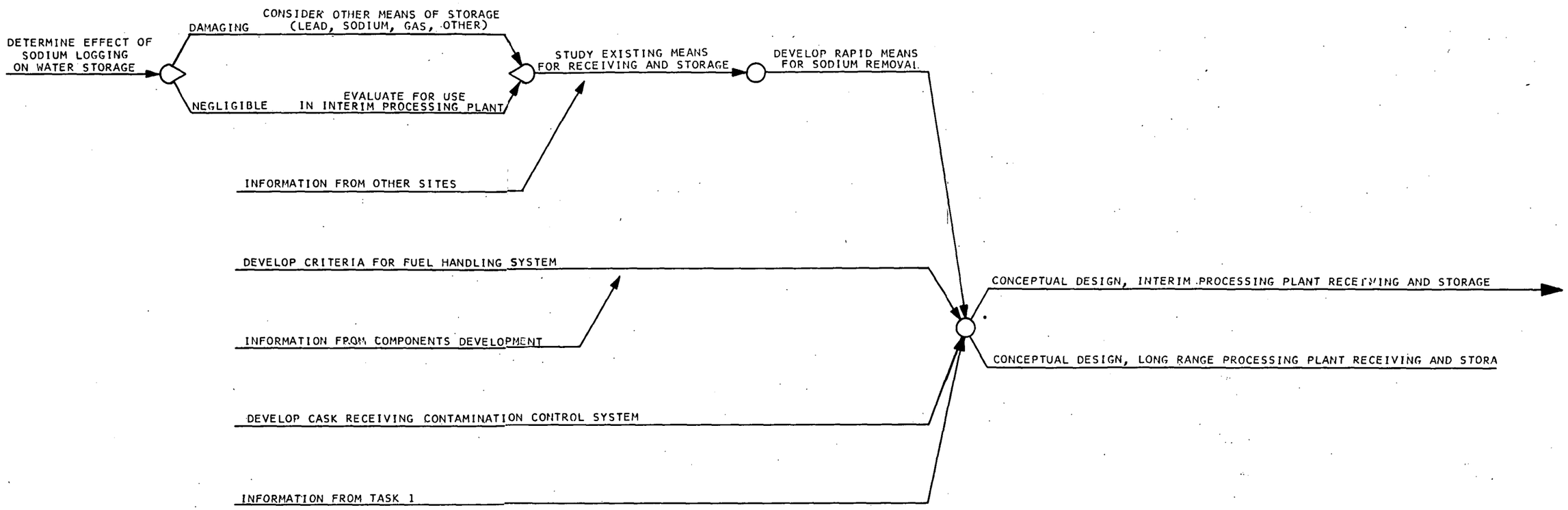

Fig. 5.5. Logic Diagram for Task 2.

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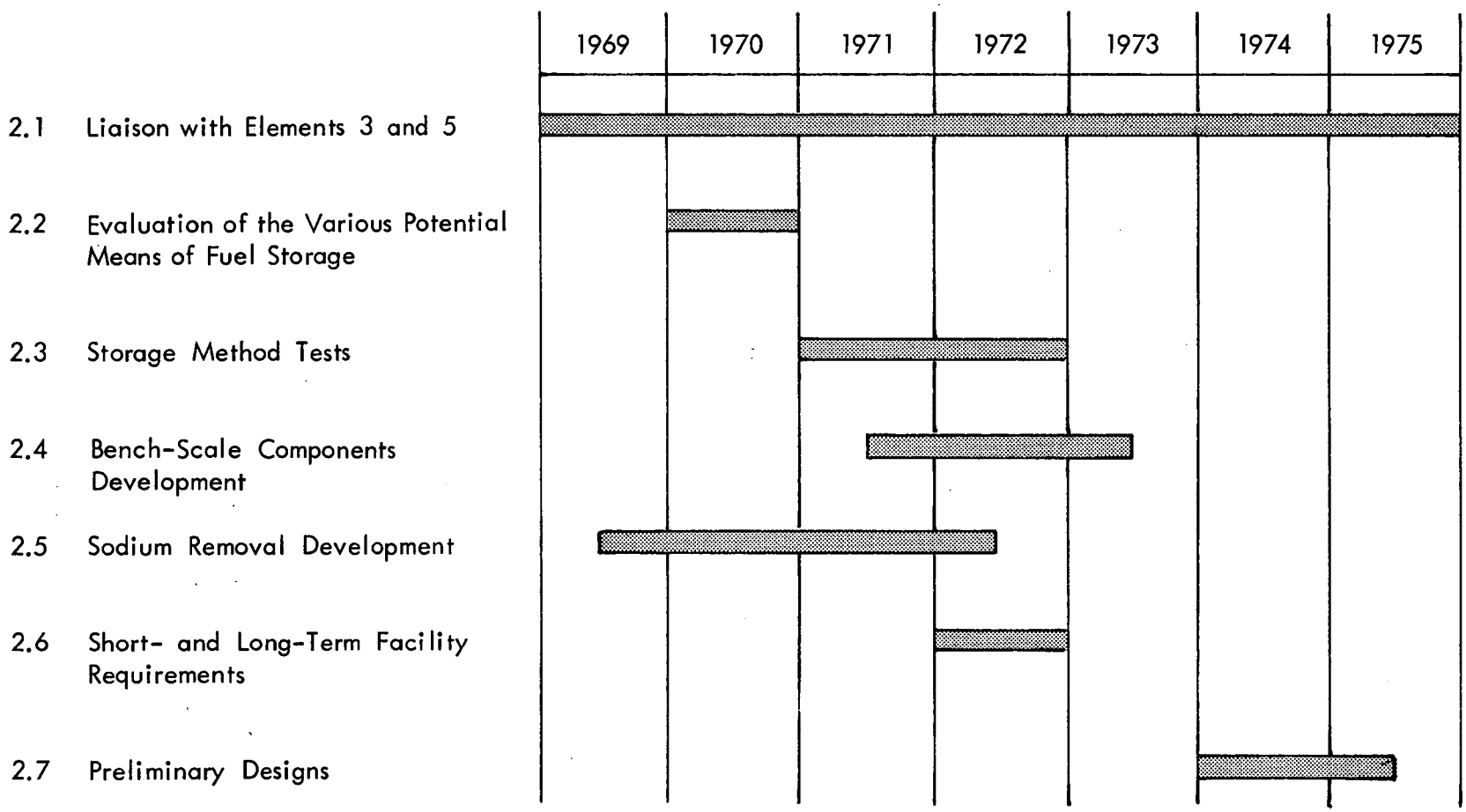

Fig. 5.6. Schedule of Activities for Task 2.

Task 2.2 Evaluation of the Various Potential Means of Fuel Storage

The various methods of fuel storage to be evaluated include, but are not necessarily limited to: (1) underwater storage, (2) storage in an inert gas-filled cell, (3) storage in an "inert" liquid (metal or otherwise), and (4) storage in sodium. The relative merits of the various methods, from the point of view of operational feasibility and total estimated cost, should be assessed.

Task 2.3 Storage Method Tests

It will likely be necessary to perform scouting tests to enlarge the scope of information on more promising storage methods, as the available information probably will prove to be inadequate for use in design of an operating facility. The work to be performed under this subtask is to provide the information that an architect-engineer would need to properly design a facility using the most attractive method(s).

Task 2.4 Bench-Scale Components Development

Many of the items necessary for the handling and storage of fuel will be developed under Task 1 of this program plan or under Element 3 of the LMFBR Program Plan. This subtask covers the development of those items not being developed under these other task areas.

Task 2.5 Sodium Removal Development

This subtask involves scaling up, and possibly expanding upon the work on sodium removal being performed under Element 5 of the LMFBR Program Plan. Methods will be developed under Element 5 (Sodium Technology) for cleaning fuel removed from the reactor as required for the inspection of fuel subassemblies and specimens exposed in reactor loops. These procedures will be directed generally toward smaller-scale and lower production rates than will be acceptable for a chemical processing facility. The work carried out under this subtask should expand upon the work carried out under Element 5 to develop equipment and methods for sodium removal that can meet the peculiar needs of the chemical processor.

Task 2.6 Short- and Long-Term Facility Requirements

In this subtask a determination should be made of requirements for the equipment and facilities needed for interim and long-range receipt and storage of fuel as indicated by the output of Tasks 1 and 3. This subtask is a part of a larger effort involving evaluation of conceptual designs for the interim and long-range processing systems. It involves a determination and weighing of the various considerations to be satisfied in the design of the two systems.

Task 2.7 Preliminary Designs

This subtask involves the preparation of a preliminary (or conceptual) design for both the interim and long-range facilities based upon the information accumulated in the other subtasks. The design should attempt only to present an outline of the desired makeup of these two facilities and should not attempt to present the basis for a construction project.

5.3 Head-End Processing - Task 3

The objective of Task 3 is to develop, evaluate, and provide design data for feasible head-end processing steps by which short- and long-decayed LMFBR fuels can be recovered using aqueous processing methods. The principal processing steps and alternatives to be developed and evaluated in this task are presented in Fig. 5.7. A logic diagram of the proposed work is given in Fig. 5.8, and a schedule of activities is shown in Fig. 5.9.

The Task 3 program includes a broad chemical and engineering analysis of the various aspects of the head-end steps and alternatives to define the most promising paths. This study would be appropriately followed by bench- and engineering-scale experimental work to obtain processing data for the scale-up and testing of reliable equipment components necessary for the design and operation of commercial processing equipment.

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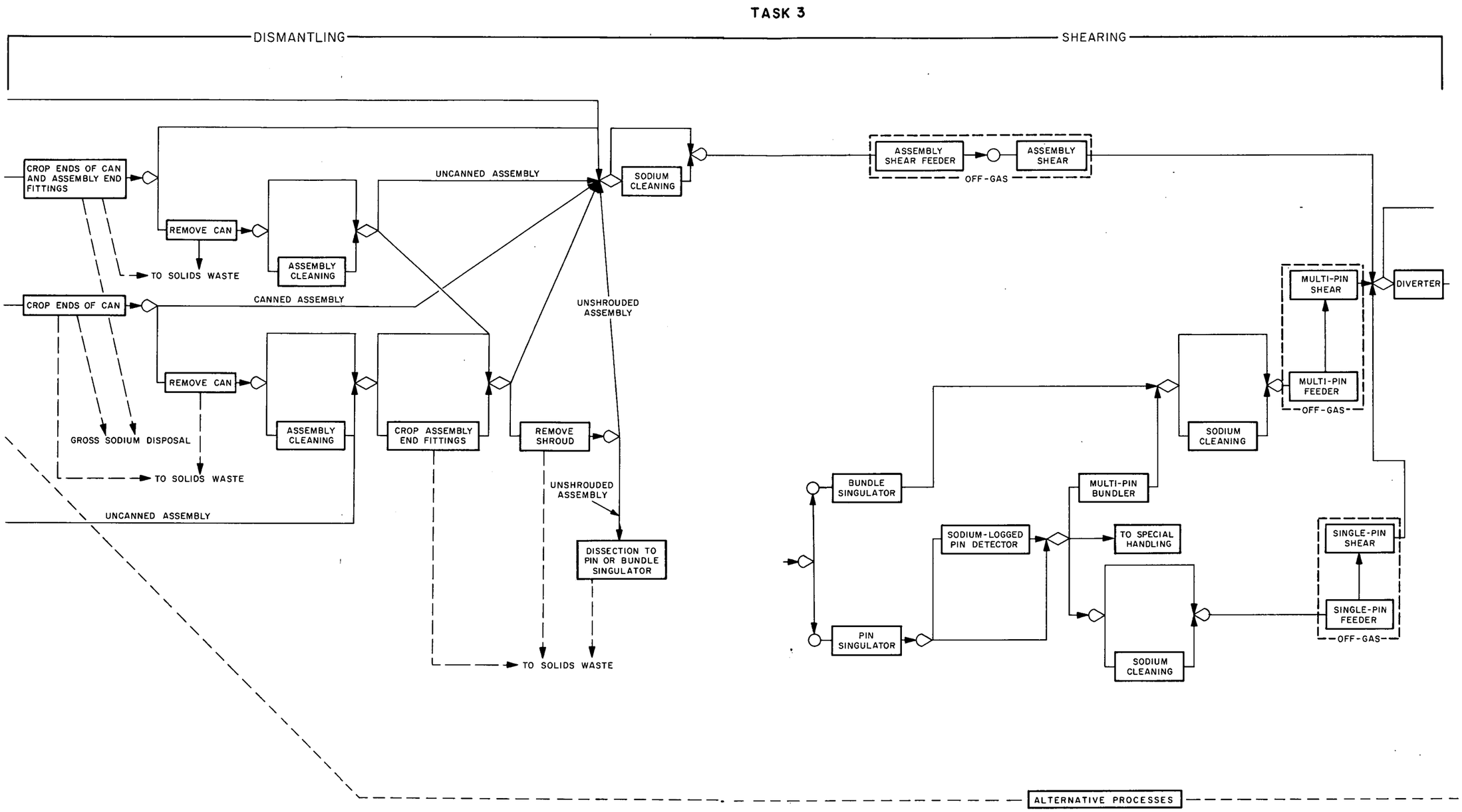

Fig. 5.7. Process Options for Task 3.

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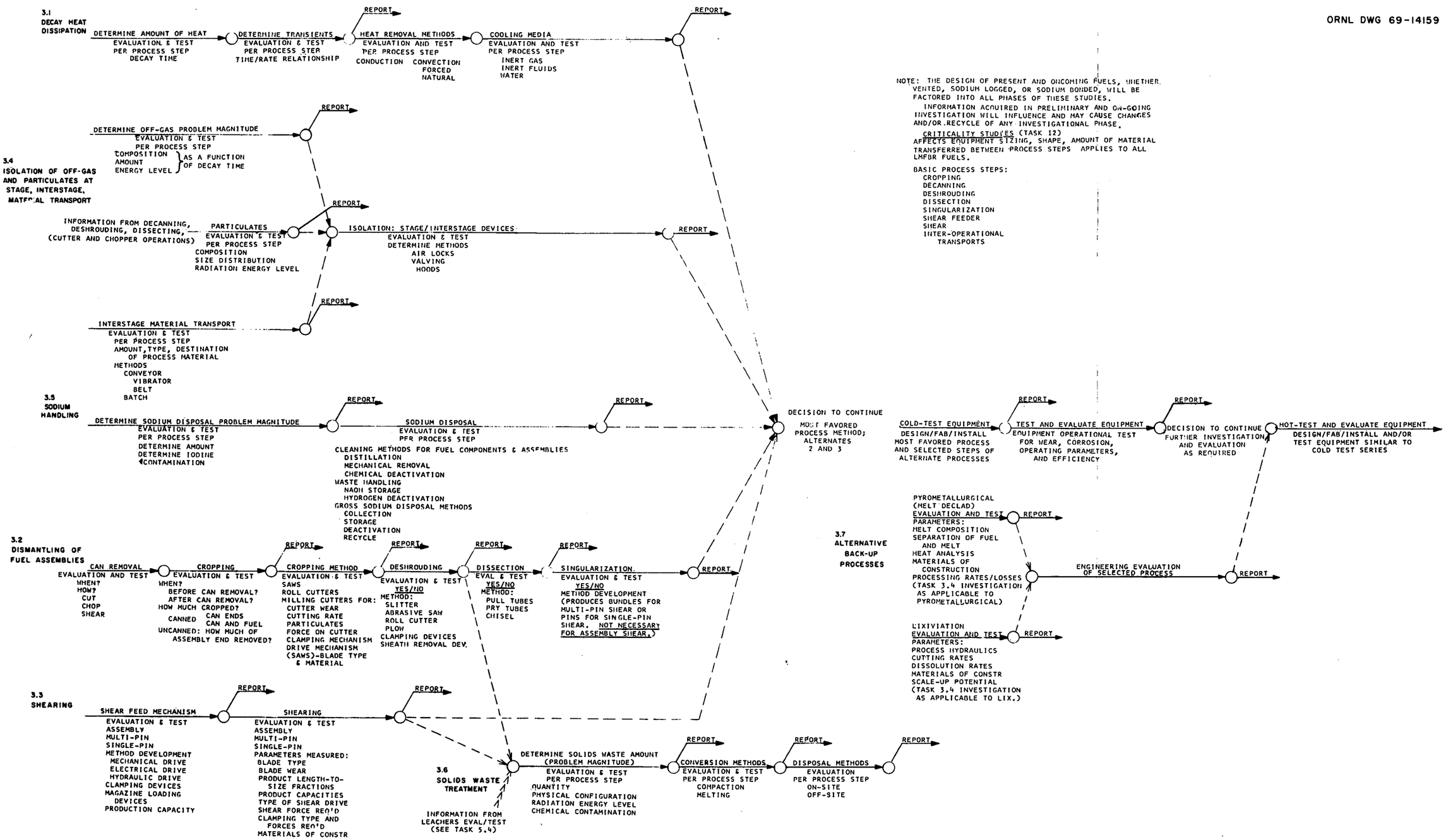

Fig. 5.8. Logic Diagram for Task 3.

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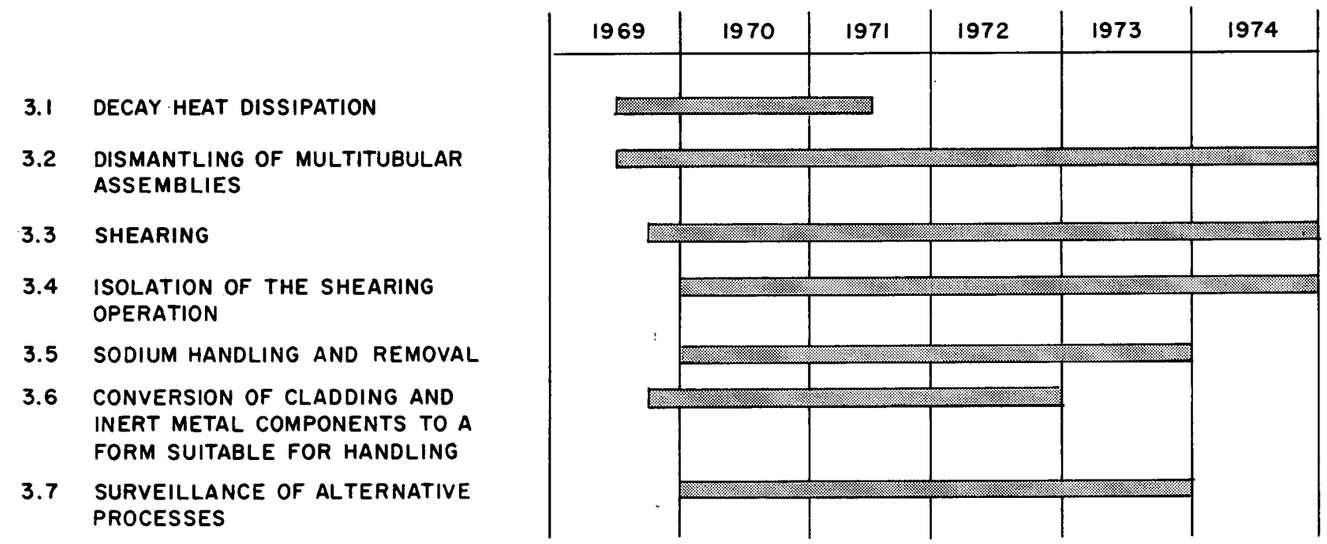

Fig. 5.9. Schedule of Activities for Task 3.

Task 3.1 Decay Heat Dissipation During Mechanical Processing

This task involves an analysis of the heat release from a broad spectrum of LMFBR fuels, such as the FFTF, GE, AI, and follow-on study fuels, and the definition and investigation of the processing limitations and requirements imposed by heat emission.

One of the purposes of the heat transfer analysis is to develop criteria for determining the conditions under which auxiliary cooling will be required for each of the head-end steps. The analysis for each step can consist of as many as three or more parts. First, the steady-state temperatures of the intact bundle or array of fuel rods of interest are found for the range of heat outputs of concern. Second, if the steady-state calculations show that the maximum allowable cladding temperature of any rod (or pin) is exceeded, transient calculations are made to determine the possibility of carrying out the operation in a safe time. Third, if auxiliary cooling is necessary, requirements such as rate of flow of cooling gas can be fixed. The results of the calculations will necessarily be dependent upon the geometrical characteristics of the array, physical properties of the fuel and cladding, and the specific heat generation rate (kw/ft) of the fuel rods. Geometrical characteristics of an array include the number of rods, its shape (circular, hexagonal, or square), the spacing of fuel rods (rectangular or triangular), pitch-to-diameter ratio, and the length and diameter of fuel rods.

The heat dissipation analyses would include decanning, cropping, deshrouding, dissection, singularization, shearing, and the alternative head-end methods. The approach taken would be similar to that illustrated in Sect. 4.3.2 but supplemented with more sophisticated mathematical techniques as new reactor design data become available. In addition, current calculations involve assumptions such as emissivities, convective heat-transfer coefficients, etc., which must be confirmed experimentally to establish the validity of the calculations. As the program develops, a new three-dimensional computer code would be desirable to replace the present two-dimensional codes. This new routine would

permit the calculation of steady-state and transient temperature profiles for nonuniform heat fluxes in the axial and radial directions for all LMFBR fuel geometries.

Task 3.2 Dismantling of Multitubular Assemblies

The objectives of this task are to study and to analyze the mechanical operations and equipment requirements involved in the dismantling of LMFBR fuel in preparation for the shearing step or other head-end operations. This task is very important because many of the downstream processing step options being considered require dismantled fuel.

Some of the principal mechanical operations to be considered are cropping, deshrouding, dissection, singularization, and shearing. The program would include the investigation, development, evaluation, feasibility of scale-up, and design data necessary for commercial applications. Deshrouding may become a very important step in the recovery of LMFBR fuel. Shroud removal is necessary for single and multirod processing and eliminates the possibility of sheared product jamming in the discharge chute of the shear.

Task 3.3 Shearing

This task would encompass a program to establish the feasibility of shearing spent fuel as cropped assemblies in a canned or uncanned condition, as multirod arrays or as single rods. This important task would also include a study of process variables, equipment requirements and limitations, and the design and evaluation of the required shearing mechanisms.

Substantial quantities of prototype fuel of the proper physical dimensions and geometric characteristics representative of all LMFBR fuels will be necessary to accomplish the program objectives. This would involve the design and procurement of prototype assemblies with the characteristics of reactor fuel but much cheaper in cost.

Task 3.4 Isolation of the Shearing Operation

The objectives of this task are to study, to evaluate, and to establish the feasibility of sealing off the shear while fuel is being sheared. The primary purpose of isolating the shearing operation is to confine particulate matter and fission gases within the shear while preventing detrimental quantities of atmospheric gas from leaking into it. Outleakage would contaminate the hot-cell facility, contribute to fuel losses, and interfere with the feeding operations and voloxidizer step. Uncontrolled inleakage would interfere in the removal and collection of fission gases evolved during shearing.

The task would encompass an analysis of valving and sealing mechanisms and techniques that would allow the passage of unsheared fuel in its various forms and the discharge of sheared fuel either as chunks or discrete pieces. Selected mechanisms would be designed and tested, and their performance would be verified in units capable of being scaled up to full contemporary and future plant requirements.

Task 3.5 Sodium Handling and Removal

The objectives of this task are (1) to establish and define the requirements and restrictions imposed on the head-end processing steps and equipment by canned or uncanned fuels that are bonded or logged internally with sodium and soiled by external adherent sodium, and (2) to provide a practical conceptual design of hot-cell equipment for deactivating and removing sodium deactivation products from intact and failed spent fuel rods and subassemblies.

The sodium cleaning or removal concept must be sufficiently versatile to permit rapid mechanical and chemical processing of small amounts of sodium, as well as gross amounts of sodium in which fuel may be immersed during shipping and storage. Such a method may not necessarily be useful in cleaning fuel specimens to be subjected to metallurgical examination.

The scope of the program would include a comprehensive study and an analysis of existing data and experience, to date, with regard to sodium removal methods, including methods such as those used in NaK deactivation

and disposal. In the analysis, available information from ORNL, the EBR II operation, Atomics International, PNL, ICPP, BNL, Savannah River, ANL, and the LMEC would be examined. The solution to the sodium deactivation problem would be approached by: (1) a wet method involving the use of moisture in an inert carrier gas, and (2) a dry method utilizing an inert carrier or sweep gas and sufficient heat to slowly volatilize sodium (but not sufficient to cause sensitization). The analysis would include all of the aspects of the safe handling, removal, and disposal of sodium, as well as other liquid metals, coolants, or liquids. Sodium would be considered first; then, other materials would be considered as their possible usage becomes important.

The scope of the program would also include some scoping-type, bench-scale experiments (or tests) to verify sodium removal parameters and the conceptual design of the hot-cell equipment. In addition, intermediate and larger-scale experiments would be made, as required, to supplement the programs at PNL, HFEF, and the LMEC; however, these experiments would be designed to meet the needs peculiar to the head-end processing steps.

A recent scoping type of survey of the state-of-the-art literature on sodium deactivation and disposal shows a lack of chemical and mechanical technology for dealing with sodium located inside and outside fuel rods or assemblies. Thus we plan to direct our attention to these areas, but only as they uniquely apply to head-end technology.

Task 3.6 Conversion of Cladding and Inert Metal Components to a Form Suitable for Handling

The objectives of this task are to study and to define feasible methods for converting the inert waste metal generated during the processing steps to a form more amenable for handling and for waste disposal. The conversion step would include an analysis and evaluation of methods such as melting, compaction, etc., to reduce the cans, end adapters, grids, hulls, wire spacers, and reflector rods to a minimal practical volume.

The initial part of the experimental program can probably be accomplished by consultation with commercial sources versed in metal reduction

methods. However, once attractive methods are defined, small-scale work with unirradiated and irradiated materials would be needed to determine the extent of off-gas and slagging problems. Also, calculations of the induced activity would be required (see Sect. 3) to determine heating effects and shield thicknesses of shipping containers for the disposal of the treated material.

Task 3.7 Surveillance of Alternative Processes

The objective of this task is to periodically survey the developmental progress of head-end steps, other than mechanical, that would be useful in recovering LMFBR fuel. The principal head-end processes of interest are:

(1) Molten metal decladding - ANL, EURATOM
(2) Melt decladding - ANL, Russian
(3) Electrolytic - Savannah River, ICPP
(4) Electrolytic-lixiviation - Eurochemic

The purpose in maintaining an interest in the processes just enumerated is to provide appropriate backup methods in the event that unforeseen obstacles appear in the mechanical head-end.

5.4 Volatile Fission Product Removal - Task 4

The objective of Task 4 is to develop an economic head-end processing method that will volatilize fission product gases (iodine, xenon, krypton, and tritium) from fuel prior to the aqueous recovery steps. This is an important task requiring the development and pioneering of a new technology.

The problem areas that would be investigated are presented as process options in Fig. 5.10 and as a logic diagram in Fig. 5.11. A schedule of activities is presented in Fig. 5.12. First, it is planned to investigate the basic chemistry of the mechanisms controlling the release of fission product gases. This would include a wide variety of irradiated fuel types (oxide, carbide, nitride) and uranium-to-plutonium

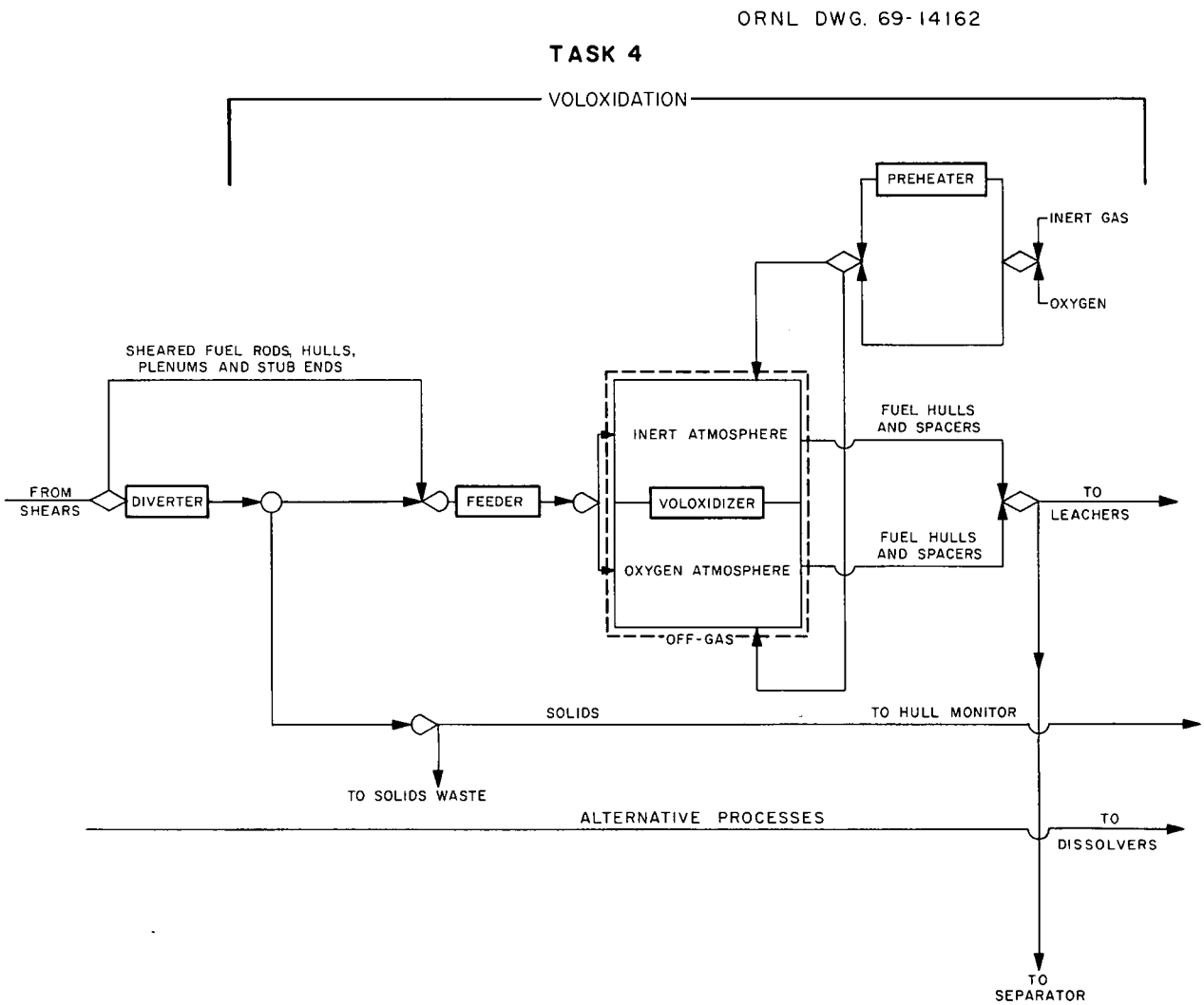

Fig. 5.10. Process Options for Task 4.

ORNL DWG 69-14161

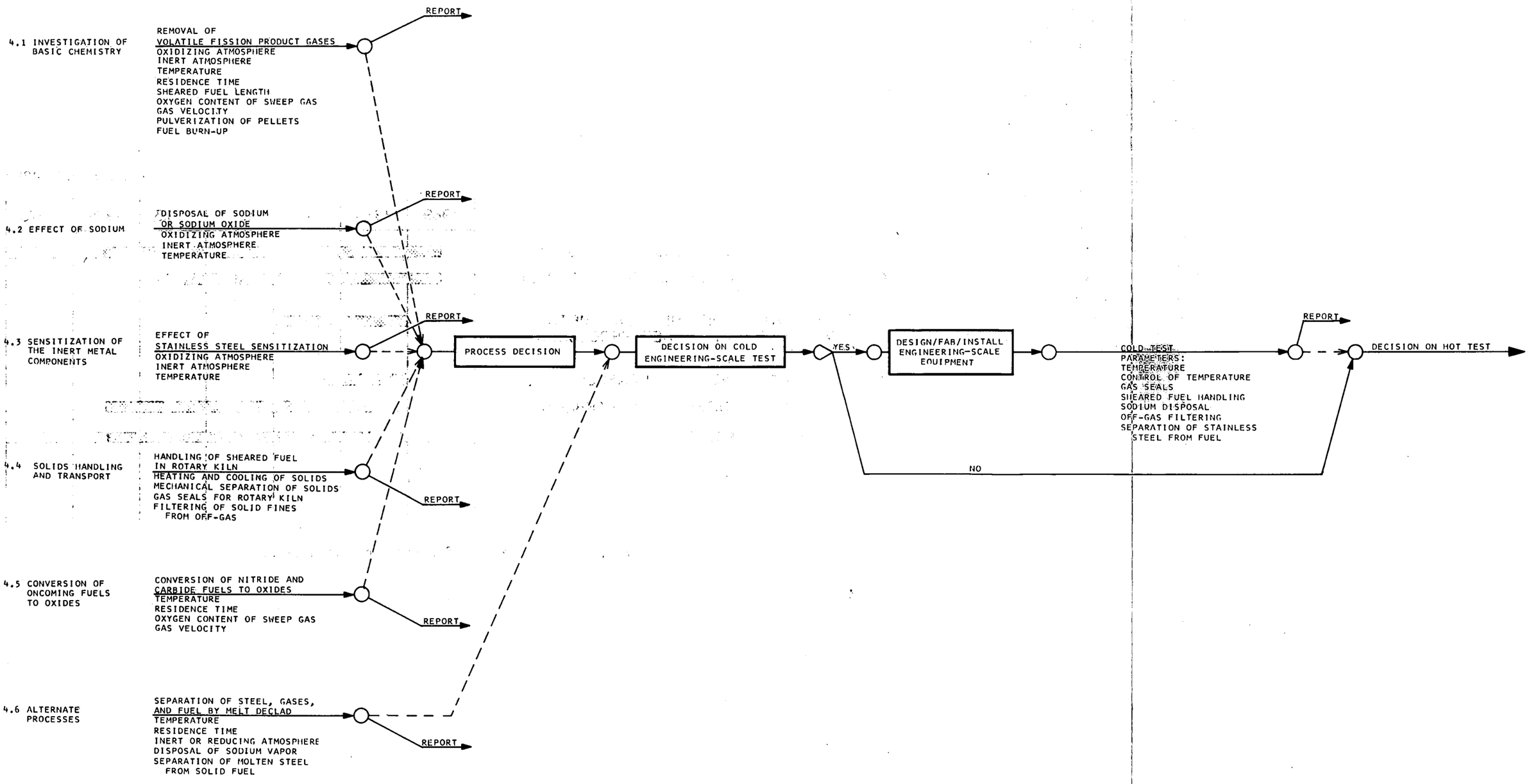

Fig. 5.11. Logic Diagram for Task 4.

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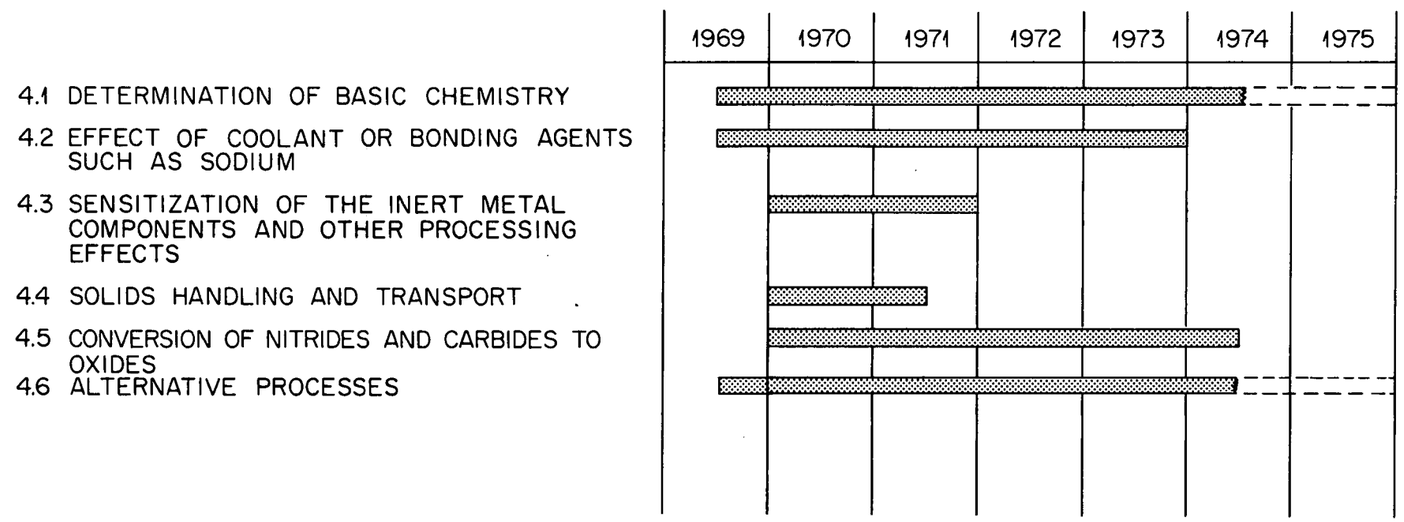

Fig. 5.12. Schedule of Activities for Task 4.

ratios. Second, as the chemistry is developed with unirradiated and irradiated fuel on a bench scale, small equipment components of various suitable designs would be evaluated with unirradiated fuels. Later, as promising types of equipment emerge, tests on an engineering scale would be desirable with unirradiated stand-in fuels. These tests would logically be followed by confirmation tests made with irradiated fuel as it becomes available.

Task 4.1 Investigation of Basic Chemistry

The purpose of this task is to develop data on the basic chemical mechanisms controlling the release of fission product gases. This would include an investigation of the reaction kinetics in an oxidative atmosphere and the effect of an oxidation-reduction cycle on the release of fission product gases. Initial studies would be performed with single pellets of unirradiated UO_2-PuO_2 and with UO_2-CeO_2 as a possible stand-in for urania-plutonia. Thermogravimetric studies would give rate data on the conversion of UO_2 to U_3O_8 and delineate the importance of crushing as an aid to oxidation. A range of pellet compositions would be investigated, for example, 10, 15, 20, and 25% plutonia.

The controllable parameters in such a study would be the temperature, residence time, sheared fuel length, oxygen content of the sweep gas, gas velocity, and ultimately the fuel burnup. The release of fission product gases from irradiated UO_2-PuO_2 pellets should also be studied in an inert atmosphere.

Task 4.2 Effect of Coolant or Bonding Agents Such as Sodium

This task is very important since the presence of sodium during fission product gas release is an unknown factor. If oxidative heat treatment is required for gas evolution, the safe destruction of sodium at these conditions would need to be measured over a range of temperatures. This would also include a determination of the competing side reactions of sodium, such as with iodine. Similarly, if an inert atmosphere is used in conjunction with heat treatment, the iodine may react almost exclusively with sodium (in the absence of O_2) and be carried as

a solid downstream to the aqueous recovery steps. Results of such effects would have to be measured, and the various reactions would have to be determined, especially in the vapor phase.

Sodium may also interact with stainless steel surfaces to produce thin films that are easily corroded in the subsequent nitric acid dissolution step. These effects, along with the possible detrimental effects of the corrosion products (and Na_2O) downstream in the solvent extraction contactors, will have to be determined experimentally.

Task 4.3 Sensitization of the Inert Metal Components of Fuel and Other Processing Effects

The objective of this task is twofold:

1. To determine whether the stainless steel hulls, spacers, and grids, and nickel reflector rods are sufficiently sensitized by the oxidative (or inert gas) heat treatment of the voloxidizer to be attacked appreciably by nitric acid dissolvent, thus incurring a significant waste penalty.

2. To define the limitations and restrictions that are imposed on the process if the operating conditions are such that the stainless steel components are prevented from becoming sensitized.

A study will be made of this problem initially on a batch scale with corrosion coupons. Later, as equipment (batch, rotary, tray, etc.) is developed, tests will be made on a larger (engineering) scale. Tests will be required in both situations in the presence and absence of sodium.

Task 4.4 Solids Handling and Transport

This task is concerned with the handling and transport of sheared fuel in the voloxidation step. The principal parameters that would be investigated include the heating and cooling of solids, filtering of fines from the off-gas system, the possible separation of stainless steel (or residues) from urania-plutonia, gas seals, and the transport of sheared material through the unit.

Task 4.5 Conversion of Oncoming Fuels (Nitrides and Carbides) to Oxides

The objective of this task is to determine the conditions required for converting the advanced fuels of uranium-plutonium carbides and nitrides to the corresponding oxides. There are several reasons, principally iodine removal and sodium destruction, for making the conversion. Also, the conversion would eliminate the presence of undesirable organic molecules formed by carbides during the nitric acid dissolution step. Nitrides are easily soluble in nitric acid, but conversion to the oxide would also safely deactivate sodium and prevent it from entering the dissolution step.

An experimental program to investigate the basic chemical reactions at various uranium-to-plutonium ratios and temperatures will be required. Equipment component development for the oxide conversion step may also be necessary. However, it is likely that equipment developed for the urania-plutonia phase change will also be useful for conversion of the nitrides and carbides.

Task 4.6 Alternative Processes

The purpose of this task is to provide suitable backup processes for shear-leach in the event that unforeseen obstacles should arise. Owing to the nature of the voloxidation step (temperature, oxidizing or inert atmosphere, etc.), melt decladding would be a logical extension of this work. Some of the principal parameters to be investigated would include residence time and temperature, separation of molten stainless steel from urania-plutonia, effect of sodium vapor, and inert versus oxidative atmosphere.

5.5 Dissolution - Task 5

The objective of Task 5 is to ensure that LMFBR fuels can be dissolved in nitric acid with high metal recoveries. Since the dissolution characteristics of the fuels can vary widely, depending on their plutonium content, method of preparation, and irradiation histories, extensive leaching

data are needed to define the effects of the many variables. A thorough understanding of iodine chemistry in the dissolver system is needed as a guide for providing effective iodine control.

The dissolver equipment must be designed and operated within rather narrow limitations imposed by criticality control and off-gas control considerations. It appears that satisfactory solutions of these problems can best be accomplished using a continuous dissolver; thus this approach should be emphasized. Evolution of a successful dissolver will require development of equipment for dependably moving the sheared stainless steel hulls and other solids through the system, and development of seals for isolating the system to prevent excessive inleakage of diluent gases.

The process options are presented in Fig. 5.13. A logic diagram of the proposed work is given in Fig. 5.14, and a proposed schedule of activities is shown in Fig. 5.15.

Task 5.1 Dissolution Data

The rate and completeness of fuel dissolution should be determined for a wide variety of oxide fuels of different compositions and preparation, temperature, and irradiation histories. Data on the weights and compositions of the leach residues and the amounts of stainless steel cladding dissolved should be obtained. On a lower priority, data should be obtained for the direct leaching of LMFBR carbide and nitride fuels and for the leaching of these fuels after a head-end treatment.

Task 5.2 Iodine Control

Extensive studies should be made of iodine chemistry in the dissolver system to identify the different iodine species present and to define their behavior under various dissolver conditions. Different potential methods of iodine control in the dissolver systems should be tested in the laboratory and confirmed in hot-cell experiments. The susceptibility of the dissolver-condenser system to corrosive attack from volatile iodine should be examined.

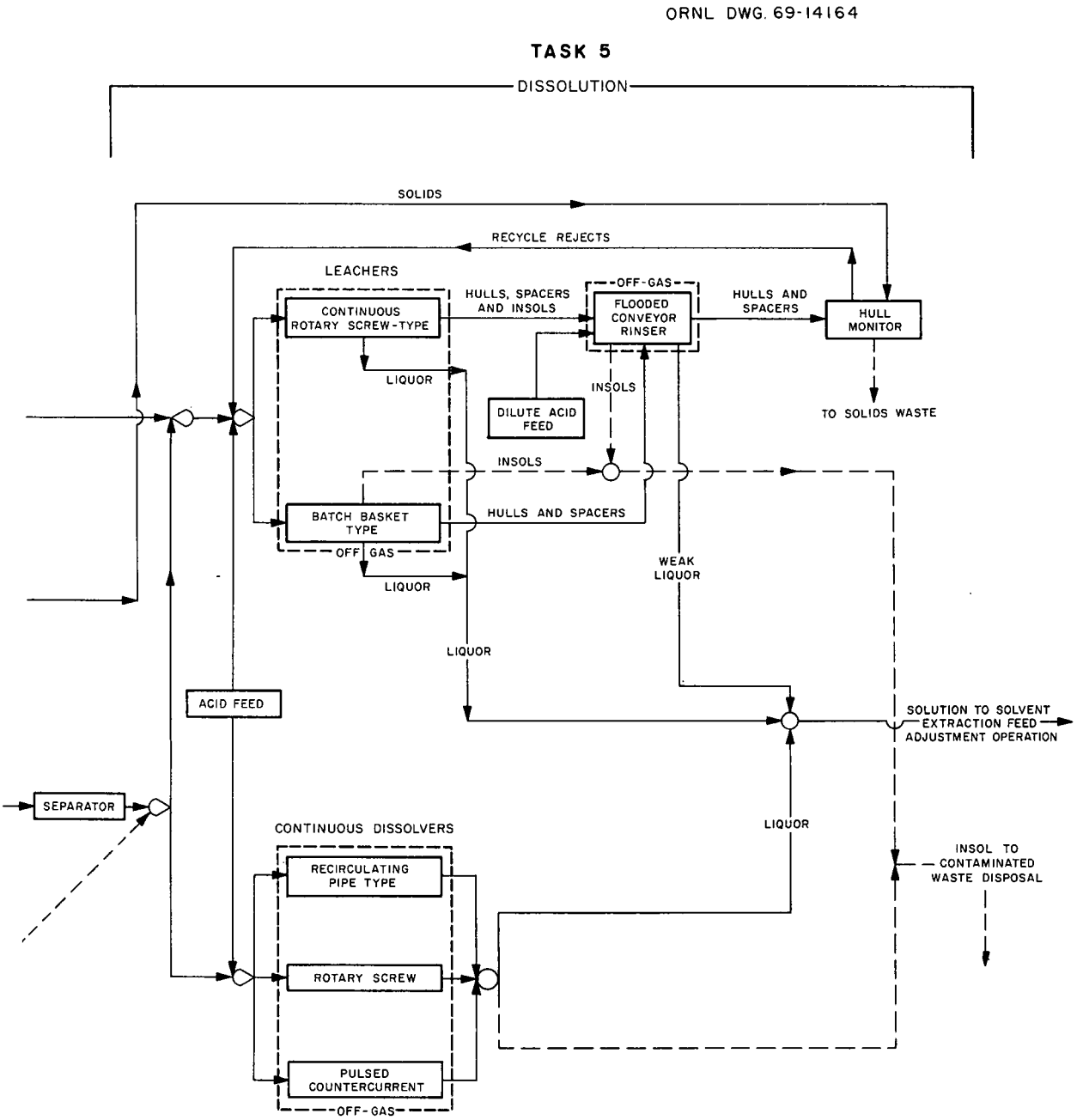

Fig. 5.13. Process Options for Task 5.

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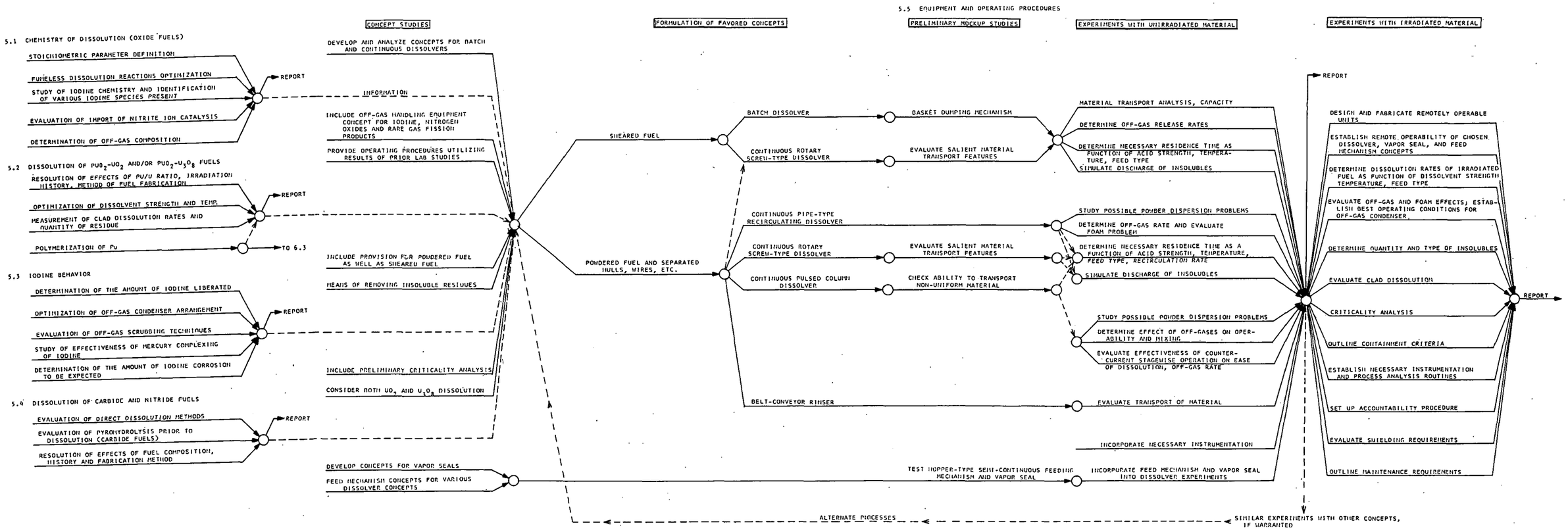

Fig. 5.14. Logic Diagram for Task 5.

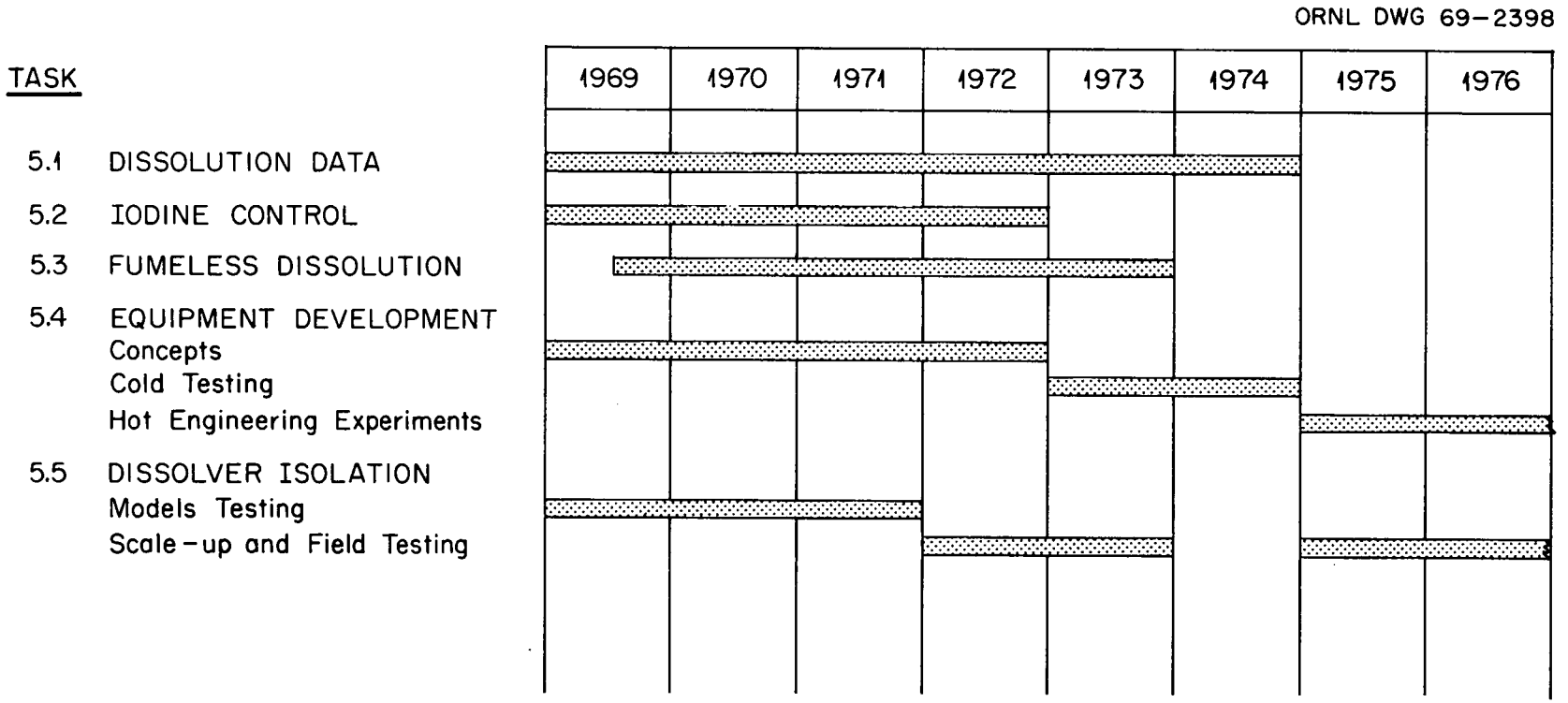

Fig. 5.15. Schedule of Activities for Task 5.

Task 5.3 Fumeless Dissolution

The use of "fumeless dissolution" techniques should be emphasized since minimizing the fume content of the off-gas will facilitate treatment of the gas for removal of iodine and noble gases. The composition of the off-gas from the dissolver-condenser should be determined for different leaching conditions, and off-gas scrubbing methods should be evaluated.

Task 5.4 Equipment Development and Testing

Various concepts of continuous dissolvers should be developed, and the most promising of these should be fabricated and tested with sheared, simulated fuel rods. Batch leaching should be evaluated as an alternative. The most promising dissolver should be tested with cold UO_2 fuels and, ultimately, in hot engineering tests with irradiated LMFBR fuels.

Task 5.5 Dissolver Isolation

A system should be developed for isolating the dissolver from the volatile fission product removal equipment (Task 4), while allowing the discharge of dissolved product and empty hulls. Mechanical, liquid, and other types of seals should be compared, and small-scale models tested. The selected method should be mocked up and field tested.

5.6 Feed Preparation - Task 6

The objective of this task is to develop methods for treating the dissolver solution to prepare it for solvent extraction. This should include studies of: (1) the adjustment of the plutonium valence, (2) removal of solid leach residues from the solution that might interfere with solvent extraction, and (3) treatment of the solution to remove iodine. Removal of essentially all of the iodine prior to solvent extraction appears necessary to alleviate problems of treating the off-gases from the solvent extraction and subsequent processing operations.

The process options are presented in Fig. 5.16; a logic diagram of the proposed work is given in Fig. 5.17; and a proposed schedule of activities is shown in Fig. 5.18.

Task 6.1 Adjustment of Plutonium Valence

This task involves the evaluation of different methods for adjusting the plutonium to the quadrivalent state, which is the form desired for most efficient extraction. The difficulty of adjusting to the Pu(IV) state and the stability of the adjusted solution should be studied as functions of the radiation density of the solution and its iron content.

Task 6.2 Removal of Iodine

Experiments should be made with both simulated and actual fuel solutions to develop efficient methods for removing iodine from the aqueous feed. Removal efficiencies should be determined for different iodine species.

Task 6.3 Feed Clarification

Solvent extraction experiments should be made with unclarified fuel solutions in order to observe the physical behavior of leach residues in the extraction system. If a need for clarification is established, evaluation of different types of clarification equipment should be made with actual fuel residues or with solids of similar physical characteristics.

5.7 Solvent Extraction - Task 7

The objective of Task 7 is to establish that LMFBR fuels can be processed successfully by solvent extraction methods. Initial emphasis should be on the development of solvent extraction flowsheets suitable for interim processing of LMFBR fuels in existing plants. Present flowsheets should be modified, where necessary, to provide for the high plutonium content of these fuels. The wide solvent extraction experience accumulated at production sites should be assessed and factored into these studies. Particular emphasis should be given to the experimental

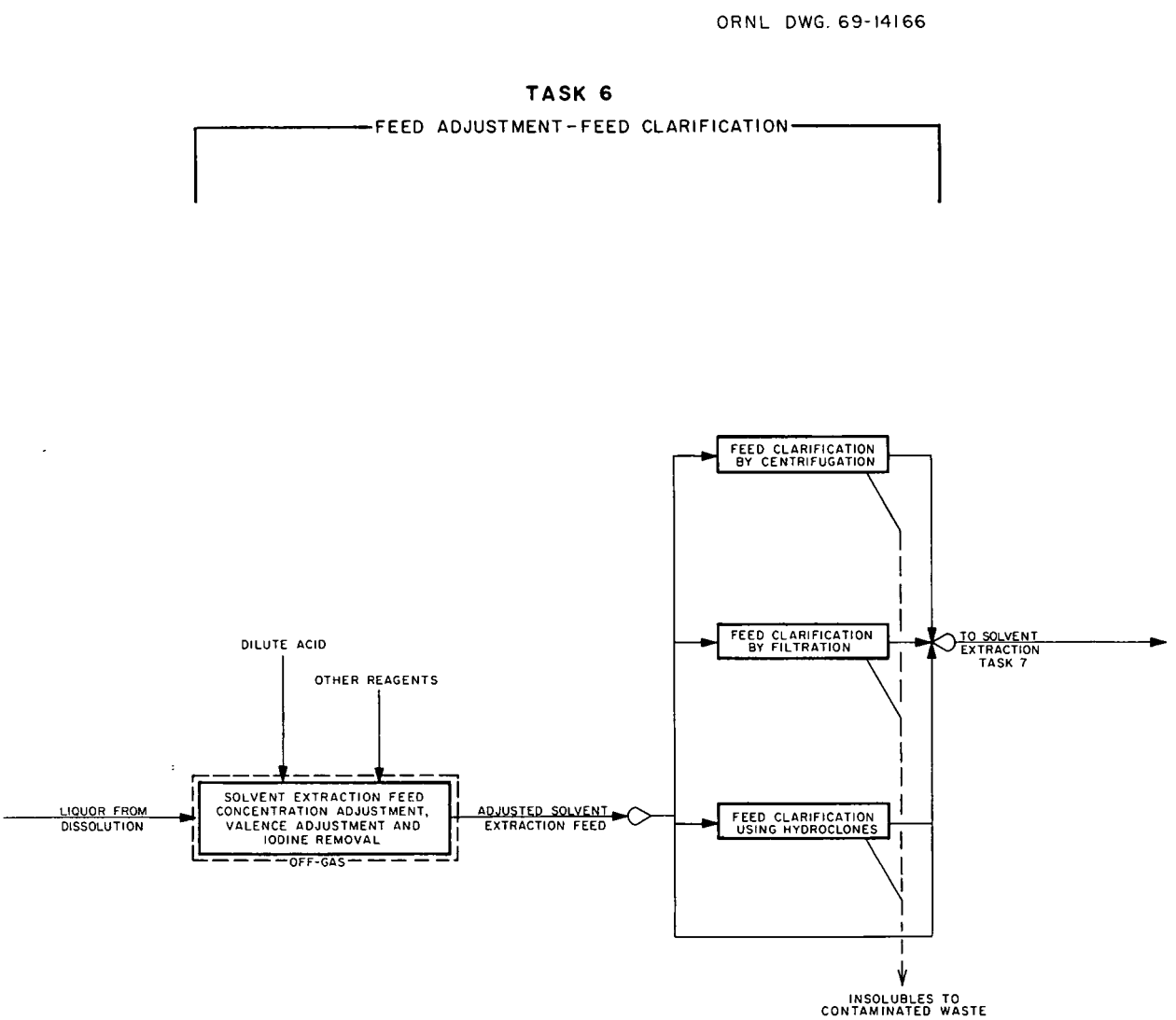

Fig. 5.16. Process Options for Task 6.

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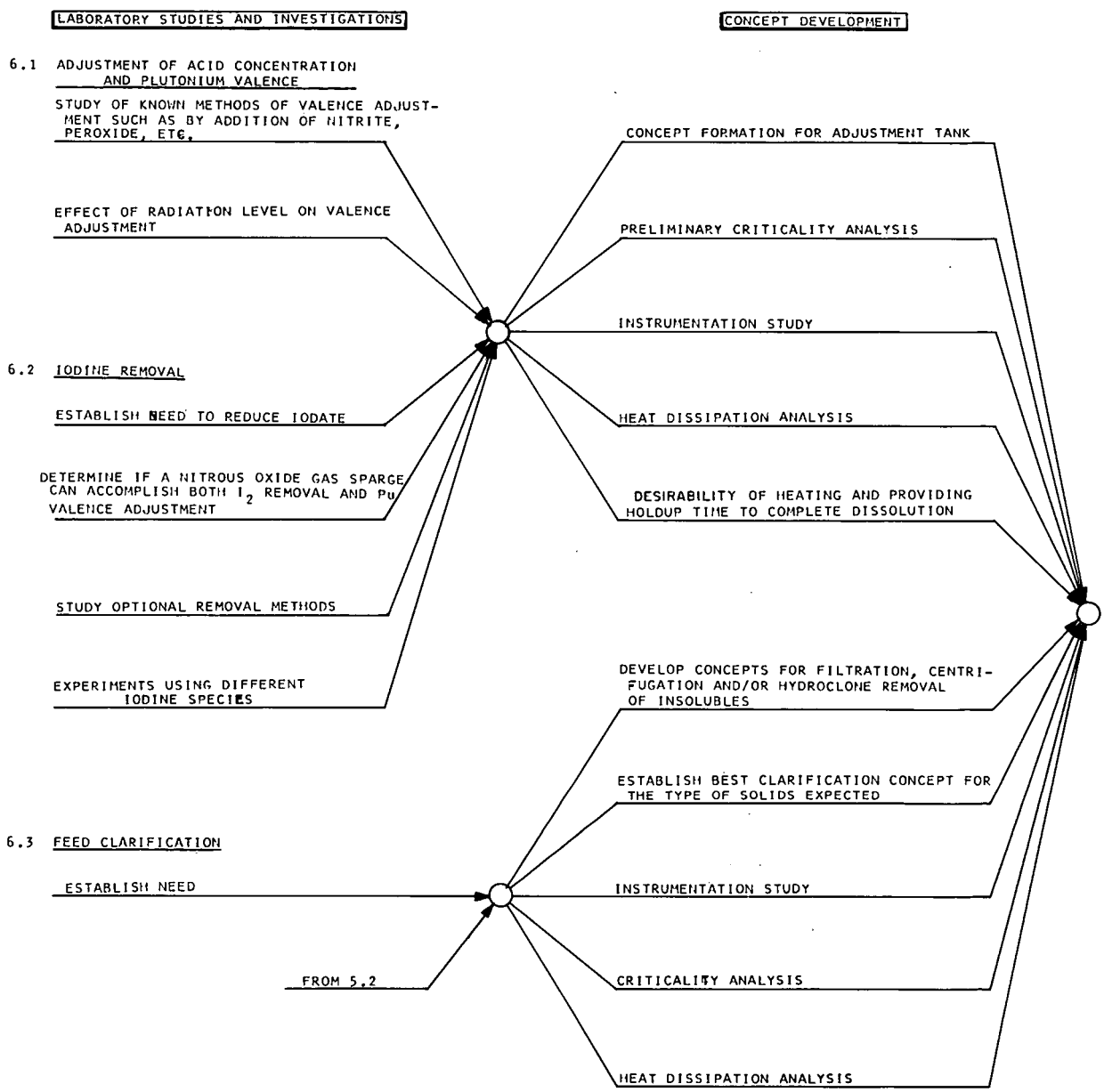

Fig. 5.17. Logic Diagram for Task 6.

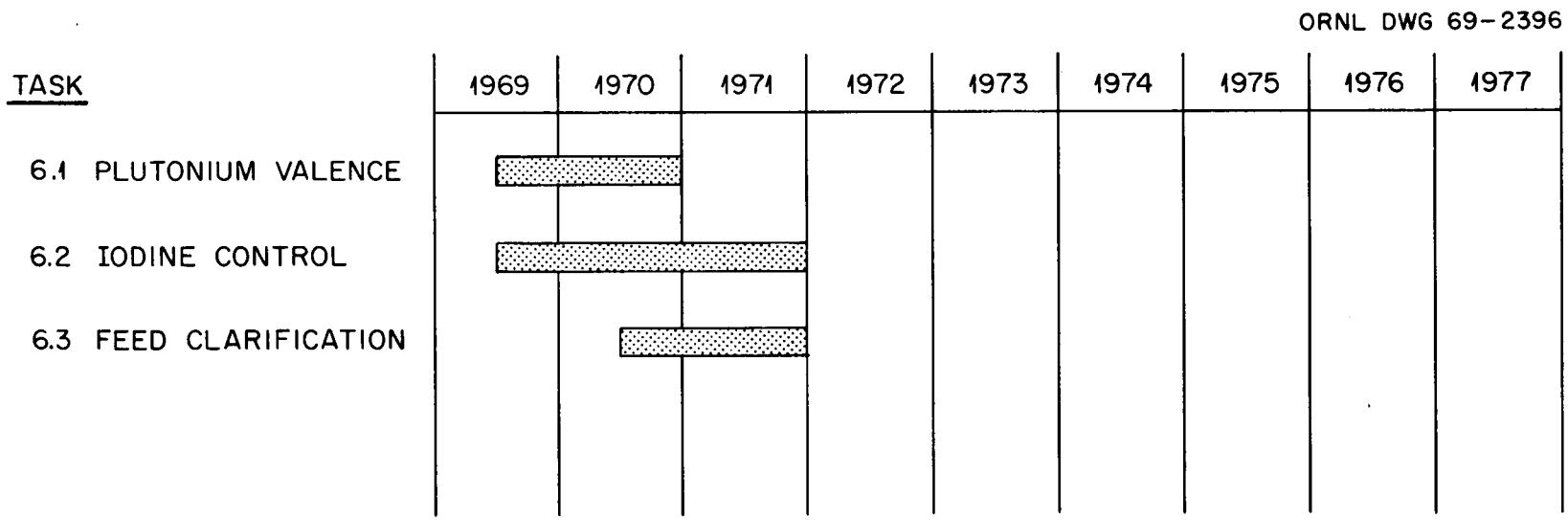

Fig. 5.18. Schedule of Activities for Task 6.

evaluation of iodine behavior in solvent extraction and the effect of solvent damage on process performance, especially when short-cooled fuels are processed. The different solvent extraction contactors should be evaluated with respect to their relative merits for processing LMFBR fuels.

The process options are presented in Fig. 5.19. A logic diagram of the proposed work is given in Fig. 5.20, and a proposed schedule of activities is shown in Fig. 5.21.

Task 7.1 Flowsheet Development and Testing

Purex flowsheets should be defined for interim processing of LMFBR fuels in existing plants. The studies should include the evaluation of different plutonium reductants in the partitioning cycle to provide a basis for choice of the optimum reductant. The flowsheets should be demonstrated in small-scale batch countercurrent systems with simulated feeds and feeds prepared by dissolving irradiated LMFBR fuel specimens. Attention should then be given to flowsheets designed specifically for processing LMFBR fuels. The flowsheets should eventually be evaluated in engineering-scale tests with highly radioactive material.

Task 7.2 Radiation and Chemical Stability of Solvent

Solvent damage effects should be studied in small-scale tests, using ^{60}Co and high-activity aqueous feeds as sources of radiation. The solvent exposure threshold level at which significant effects occur should be determined. Studies should be made to identify the offending degradation products and their concentrations as a function of radiation dose. The rate of decomposition of TBP in nitric acid should be determined for the range of process temperatures and acid concentrations expected. The conditions under which metal precipitates of DBP and MBP occur should be defined as a guide to possible difficulties arising from the precipitation of these salts in the extraction system. The importance of nitration of the diluent and of TBP should be examined. On a low-priority basis, alternative extractants should be examined and, if justified, should be evaluated experimentally for the processing of LMFBR fuels.

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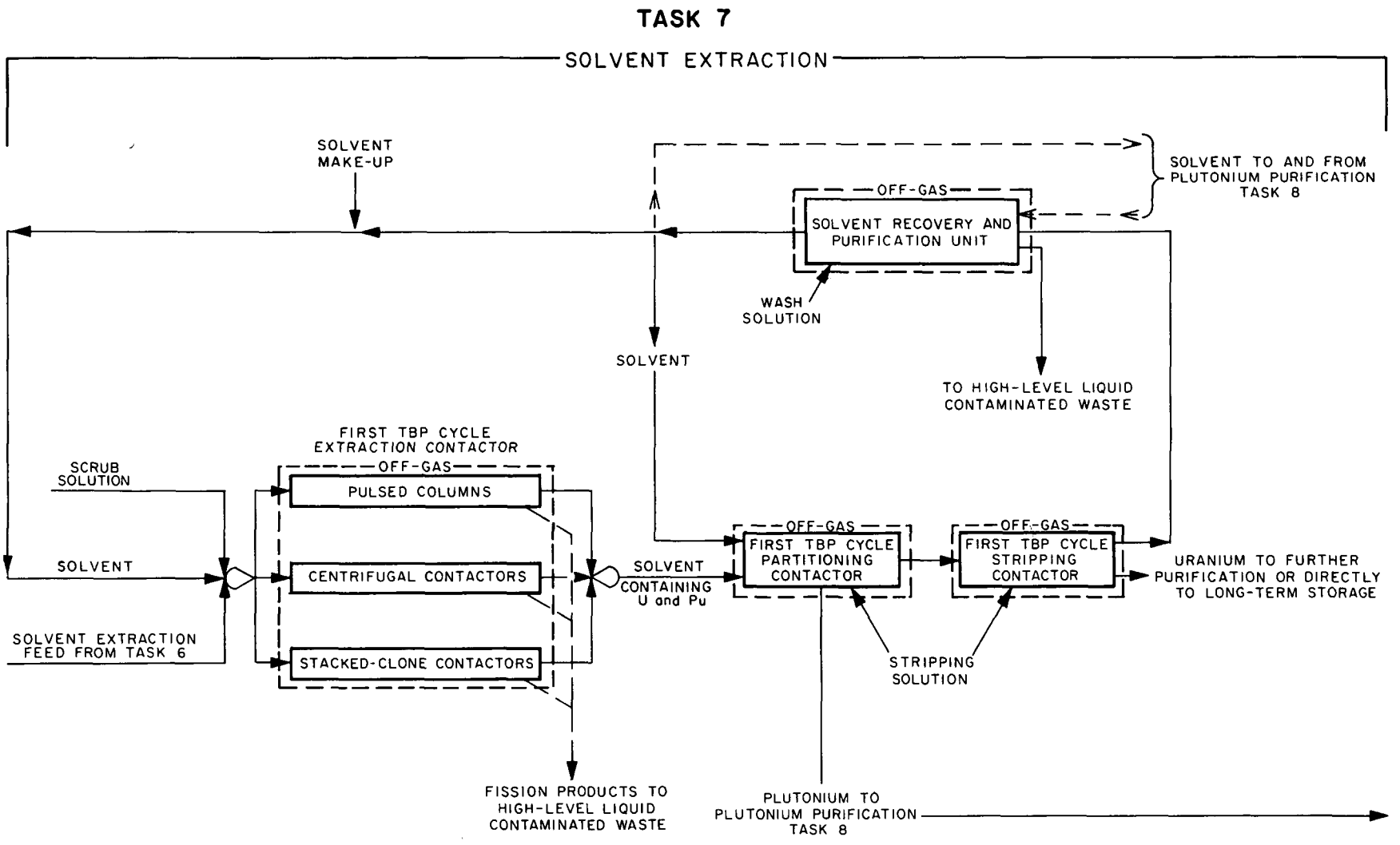

Fig. 5.19. Process Options for Task 7.

ORNL DWG 69-14167

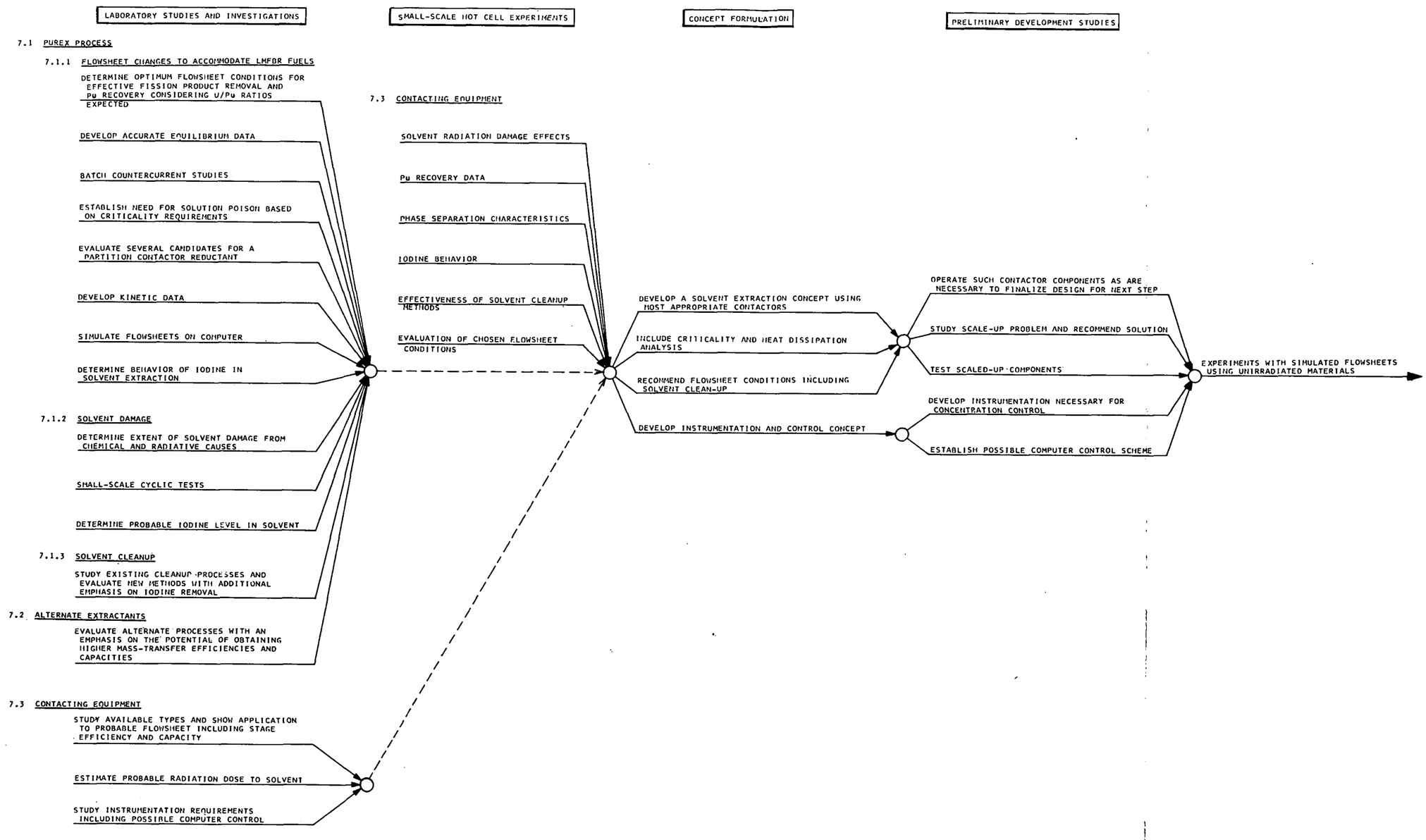

Fig. 5.20. Logic Diagram for Task 7.

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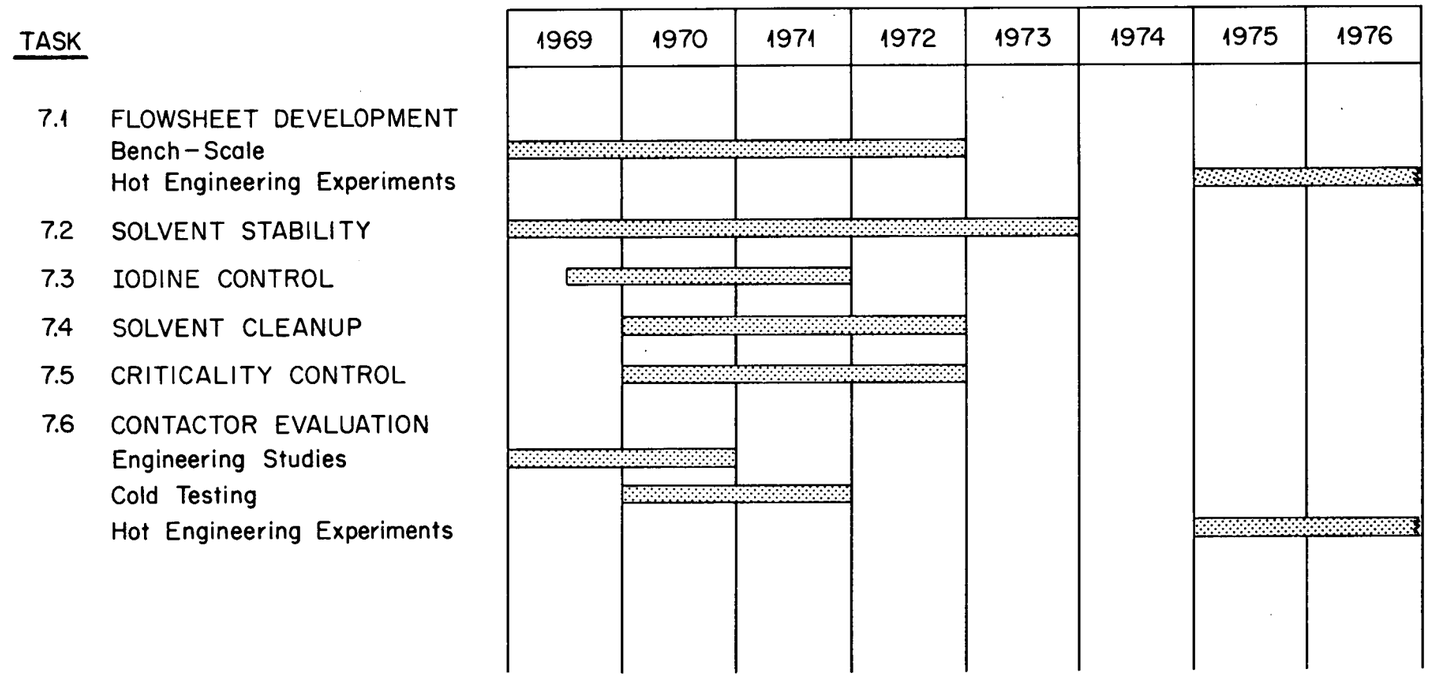

Fig. 5.21. Schedule of Activities for Task 7.

Task 7.3 Iodine Control

The extraction of the different iodine species by Purex solvent should be examined. The rate of reaction of the extracted iodine with the solvent to form organic iodides should be determined, and the products of the reaction should be characterized. Methods of stripping iodine from the solvent should be sought.

Task 7.4 Solvent Cleanup

A literature survey should be made of solvent cleanup methods, and the more favorable of these should be evaluated experimentally. If the present methods are not adequate, new methods should be developed.

Task 7.5 Criticality Control

Different methods for controlling criticality in the solvent extraction circuits should be examined with respect to their effects on the chemical flowsheet. Where concentration control is used, the flowsheets should be tested under conditions of assumed maloperation to make certain that critical concentrations would not be exceeded through excessive plutonium reflux or plutonium segregation (i.e., third-phase formation). The behavior of proposed soluble poisons in the solvent extraction circuits should be studied.

Task 7.6 Contactor Evaluation

Engineering studies should be made to compare the merits of different contactors for use in solvent extraction of LMFBR fuels. The contactors should be compared from the standpoints of: (1) solvent radiation exposure, (2) adaptability to criticality control, (3) size and cost of the contactor and the shielded facility required for various plant throughputs, (4) solvent inventory, and (5) ease of process control.

5.8 Plutonium Purification - Task 8

The objective of Task 8 is to provide methods for separating plutonium from uranium and fission products in a form having adequate purity and

concentration for delivery to the fuel refabrication system. Although the plutonium process chemistry is reasonably well-known, the higher plutonium content of LMFBR fuels makes some process revisions desirable. A need for a process with a higher capacity than ion exchange (or a need for modification of the ion exchange process or equipment) for final decontamination of plutonium is evident for LMFBR fuel processing plants.

The process options are presented in Fig. 5.22. A logic diagram for the proposed work is given in Fig. 5.23, and a proposed schedule of activities is shown in Fig. 5.24.

Task 8.1 Second TBP Cycle

Flowsheet conditions should be established for adjusting the oxidation state of the first-cycle plutonium product solution and for recovering plutonium from this solution in the second TBP cycle.

Task 8.2 Ion Exchange Processing

Based on available information, the potential applicability of anion exchange to the processing of LMFBR fuels should be critically assessed. This assessment should include possible use of small-diameter resin beads in a high-pressure column. If necessary, an experimental program should be carried out to obtain additional data needed for the assessment.

Task 8.3 Amine Extraction Process

An amine extraction process for the final purification of plutonium, using secondary amines (which can be stripped with dilute acid), should be outlined on the basis of batch tests. The process should be tested with simulated and actual feeds to demonstrate plutonium recoveries and the separations obtainable from fission products. The chemical and radiation stability of the secondary amines and the effects of solvent degradation on process performance should be determined. Throughput capacities of different contactors should be determined for the amine solvents. Methods of solvent cleanup should be developed and evaluated. Sufficient data should be obtained in these studies so that an engineering evaluation can be made of the relative merits of the amine process, as compared with ion exchange, for use in LMFBR plants.

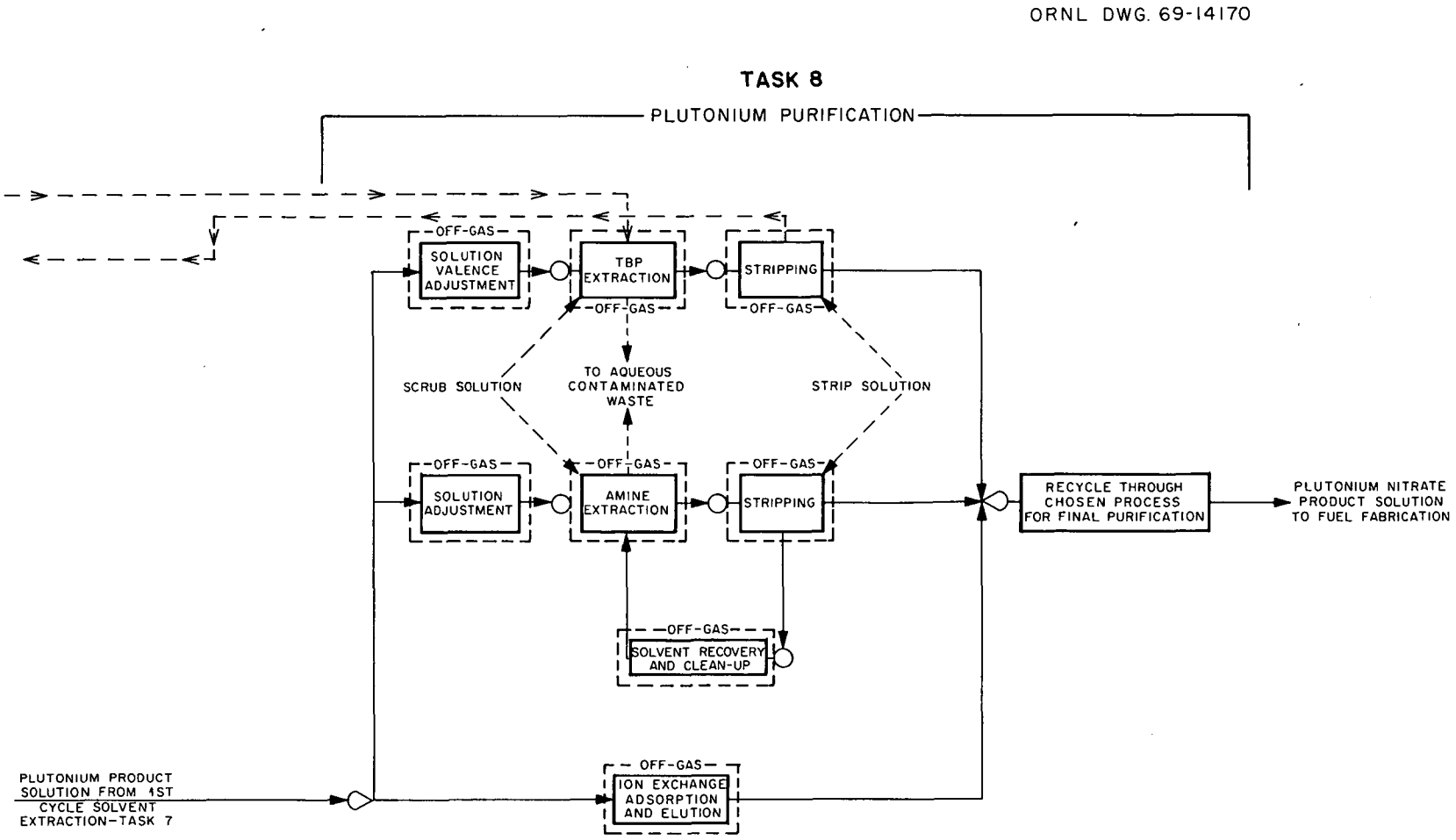

Fig. 5.22. Process Options for Task 8.

ORNL DWG 69-14169

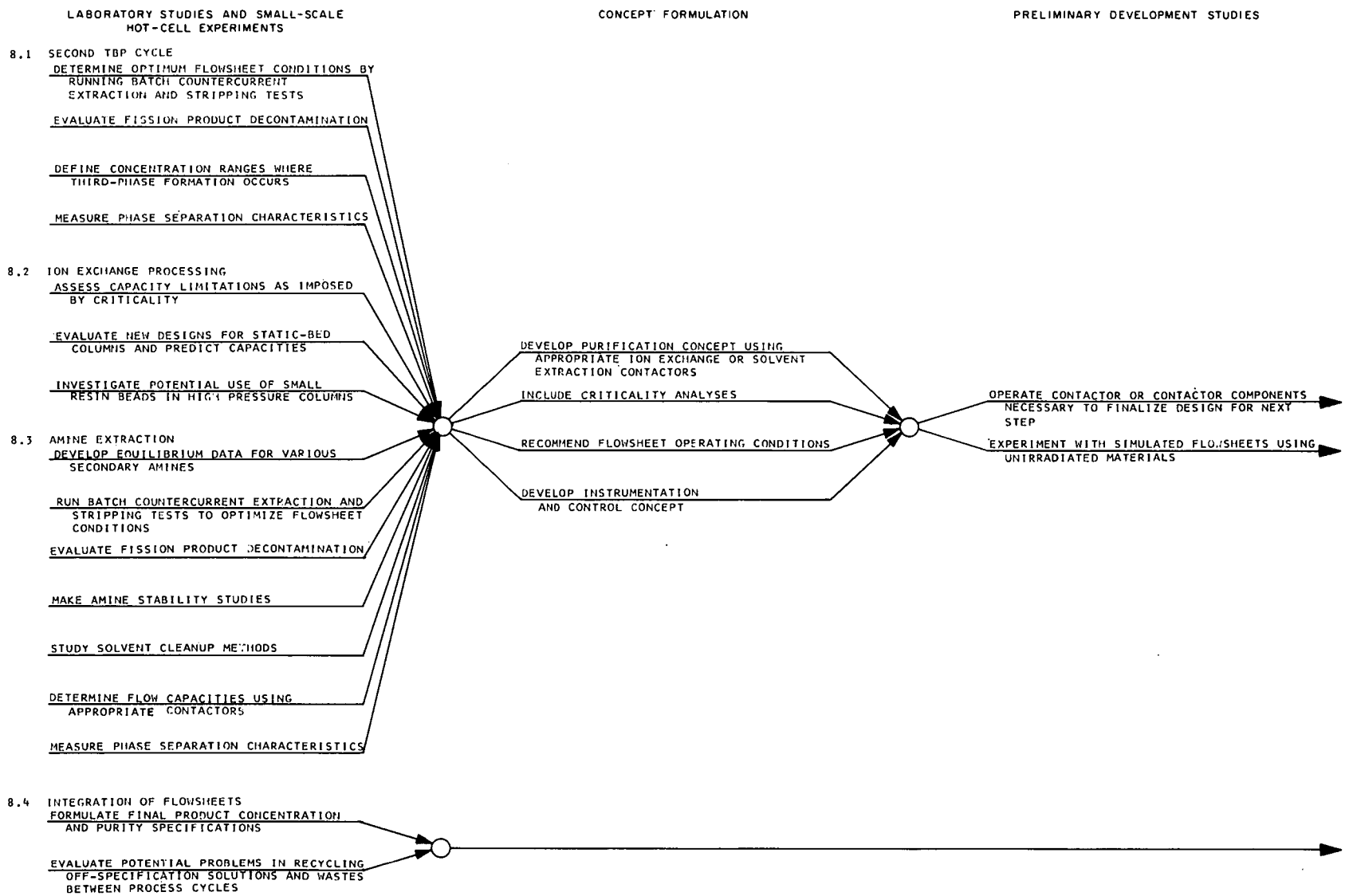

Fig. 5.23. Logic Diagram for Task 8.

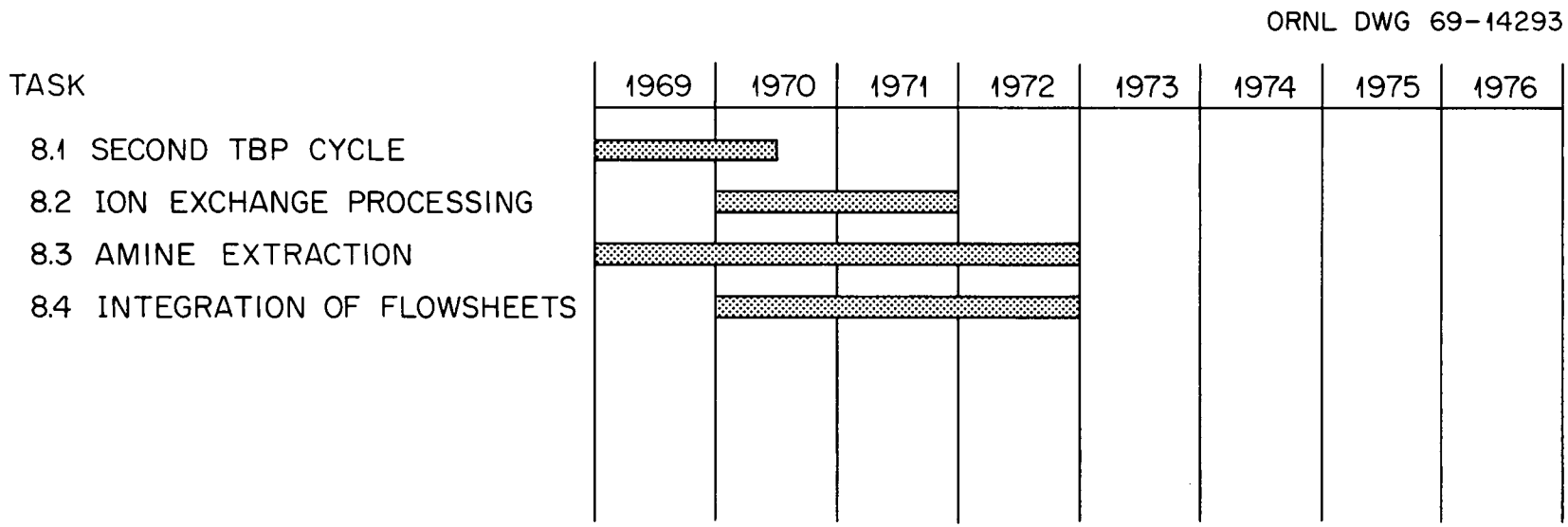

Fig. 5.24. Schedule of Activities for Task 8.

Task 8.4 Integration of Flowsheets

Different arrangements of the plutonium purification process flowsheets should be considered and evaluated with the aim of minimizing the number of purification cycles (and increasing their efficiencies), simplifying process wastes, and providing easy integration with processes for reconstituting the fuel. A third cycle of TBP extraction should be considered as an alternative to ion exchange or amine extraction for final purification of the plutonium. The laboratory studies should include the evaluation of aqueous complexing agents and scavenging treatment for improving the separations from fission products.

5.9 Waste Treatment and Storage - Task 9

The objective of the waste treatment and storage program is to develop economical processes for retention, immobilization, storage, and disposal of wastes. These methods will be used to optimize safety considerations in siting and operating fuel processing plants and final waste disposal areas.

The process options are presented in Fig. 5.25. A logic diagram of the proposed work is given in Fig. 5.26, and a proposed schedule of activities is shown in Fig. 5.27.

Task 9.1 High-Level Wastes

This task involves the development and comparative evaluation of methods for: (1) interim storage of liquid and gaseous wastes, (2) immobilization by conversion to solids, (3) interim storage of solid wastes, (4) shipment to the final disposal facility, and (5) final disposal.

LMFBR Liquid Wastes. - LMFBR liquid wastes will be similar in ionic content to LWR wastes but will have higher heat generation rates per ton of fuel at equal cooling times because of the high specific power of LMFBR fuel. A comprehensive survey has been undertaken to evaluate the various steps in management of the wastes that are expected

ORNL DWG. 69-14172

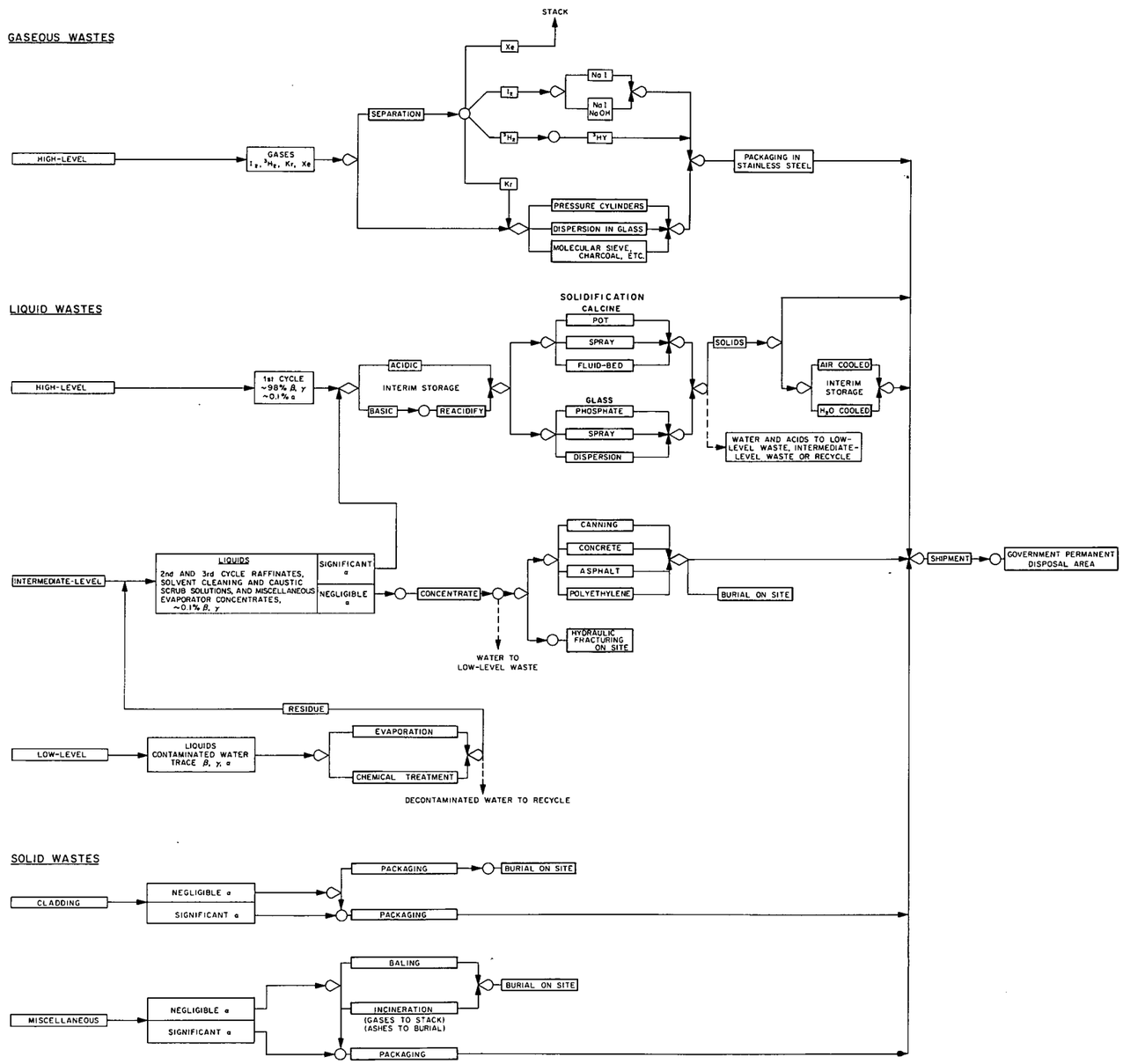

Fig. 5.25. Process Options for Task 9.

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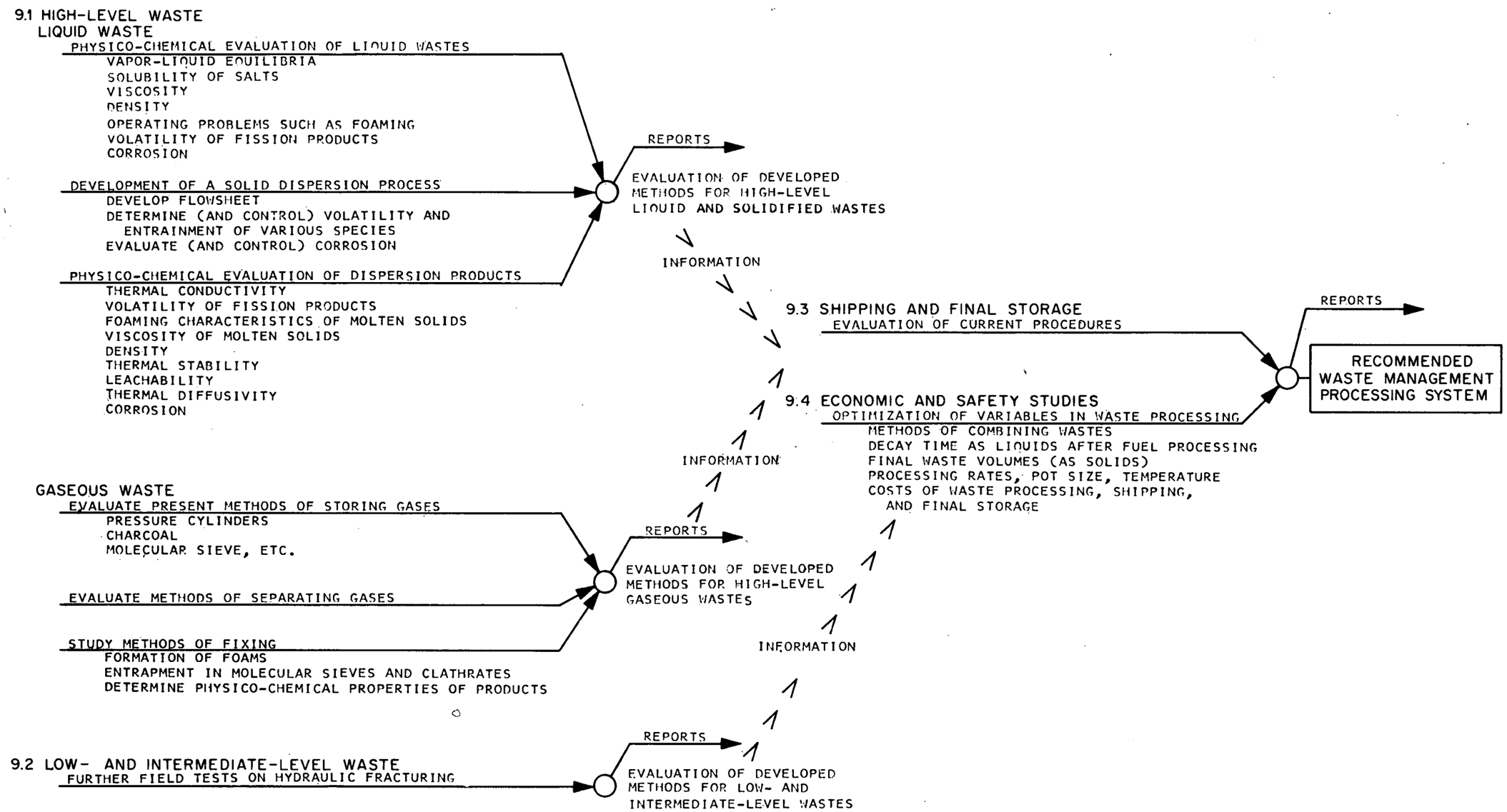

Fig. 5.26. Logic Diagram for Task 9.

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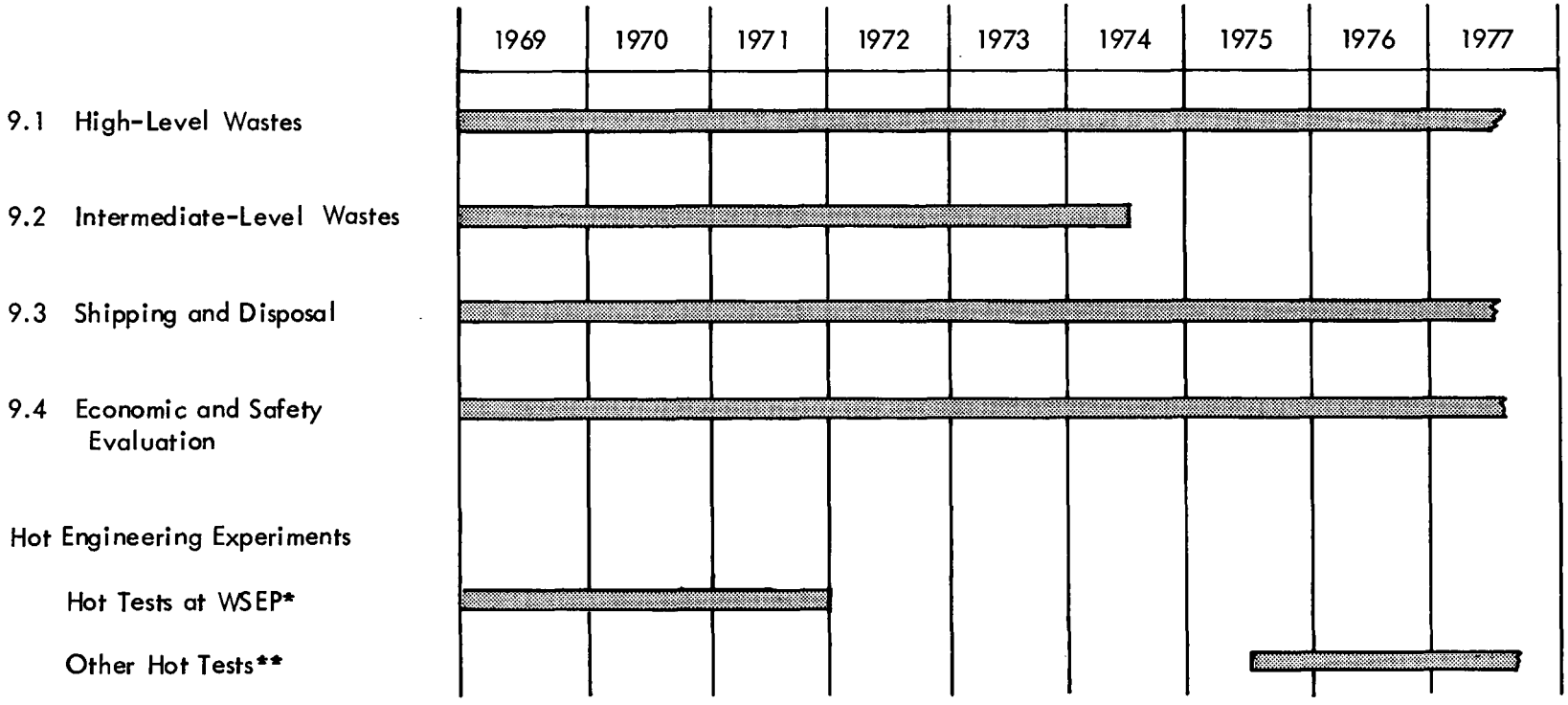

*Waste Solidification Engineering Prototype at Pacific Northwest Laboratory.

**Oak Ridge National Laboratory

Fig. 5.27. Schedule of Activities for Task 9.

to be generated as a result of the aqueous processing of fuels from LMFBR's, as well as other reactors. The survey should consider costs, operating efficiency, and safety as a means of comparing and optimizing alternative processes and operations for each step. While there is a safety incentive to solidify as early as possible, there is an economic incentive to solidify in a minimum number of containers having as large a diameter as possible. A number of properties and parameters must be properly interrelated to optimize both safety and economics. These include thermal conductivity, density, cooling time in liquid storage, number and size of pots, dilution with inert solids, the combination of various intermediate- and high-level wastes, and cost of liquid storage versus storage as solids. Developmental work is proceeding to characterize these wastes (and the solids that could be formed from them) and to provide the data required for the survey. A method of increasing the thermal conductivity of the product for ultimate disposal by dispersing the waste solids in a glassy matrix is being studied. The aim of these studies is to produce a dispersion at 900°C or less with the fission-product waste, using the sodium in caustic scrubber and solvent cleanup waste and the boron in the fission product waste (plus any additional boron required) to form a sodium tetraborate continuous glassy phase, or, with the addition of silica, to produce a borosilicate continuous glassy phase. A dispersion is preferable to a single-phase glassy or crystalline product since it has some of the desirable properties of each, namely, the lower leach rate of the glassy, continuous phase and the higher thermal conductivity of the dispersed crystalline phase.

LMFBR Gaseous Wastes. - LMFBR gaseous wastes will include I_2, ^3H, Kr, and Xe. Immediate immobilization of these gases is desirable to decrease the possibility of their sudden release in an incident. An engineering, economic, and safety survey should be made to compare alternative disposal methods. At present, the only available method is storage in pressurized cylinders. New methods proposed for the disposal of gaseous wastes include conversion of iodine (half-lives, 8.05 days and 1.6×10^7 years) and tritium to stable solid compounds for storage, and storage of krypton as a gas after separation from the inert xenon.

Xenon (half-lives, 5.27 and 12 days) can be released to the atmosphere after a few months, but krypton (half-life, 10.76 years) must be retained for long-term storage.

Iodine is expected to be retained in caustic scrubber solutions or to be transferred to caustic solutions after retention by other off-gas removal methods (see Sect. 5.10). Hence, the problems involved in solidification of caustic, and the physical and radiochemical stability of the iodine compounds in the product, should be determined. Other methods of forming stable solid products from elemental iodine collected in cold traps or other gas treatment methods should also be studied. Similarly, methods for forming stable solid compounds of tritium, such as Na^3HSO_4, require evaluation. The solid waste products could be sealed in a stainless steel primary containment vessel (pot) for additional safety during temporary storage and shipment to the government disposal area. However, both primary and secondary containment seem desirable for gases as opposed to simple pressurized tank storage. Several alternative primary containment methods should be considered; in each case, a stainless steel pot or pressure vessel should serve as the secondary containment vessel. These containment methods include entrapment in molecular sieves, dispersion as a foam in glass, plastic, or metal media, or dispersion as microampuls in similar media. Initial experimental tests indicate reasonable promise, but additional work is required to permit meaningful engineering and economic comparisons.

Task 9.2 Low- and Intermediate-Level Wastes

Processes are being developed for injecting waste-cement slurries into deep geologic formations by hydrofracturing, or for incorporating all types of intermediate-level waste residues in asphalt, polyethylene, or other plastic material prior to burial. These wastes would not contain significant amounts of long-lived alpha nuclides; they include organic wastes and the residues from: (1) second- and third-cycle solvent extraction raffinates, (2) solvent cleanup, (3) off-gas scrubbers, and (4) slurries, concentrates, or solids from chemical treatment or evaporation of low-level wastes. Wastes containing significant amounts of long-lived

alpha nuclides would be combined with high-level wastes. The incorporation of waste in asphalt and polyethylene has been carried through laboratory and engineering development, and no additional development is now planned. Further development of the hydraulic fracturing technique includes additional field tests.

Task 9.3 Shipping and Final Storage

Present plans are to establish a federally owned disposal site in a salt mine to assess the problems involved in the operation of a final disposal facility. The shipping techniques and knowledge developed in Task 1 will have direct application in this task.

Task 9.4 Economic and Safety Studies

Economic and safety studies will be made to evaluate alternative methods for the management of LMFBR wastes. A code will be derived to optimize the large number of variables to be studied in the initial survey and to facilitate subsequent studies and evaluations. Variables include volumes and solids content of high- and intermediate-level liquid wastes, alternative methods of combining wastes for storage, effect of thermal conductivity, and effect of solid waste type (porous calcine, solid glass, dispersion in glass, etc.). Initial emphasis will be placed on: (1) the effect of combining all wastes containing high concentrations of alpha emitters with the first-cycle extraction waste, (2) the importance of immobilization of gaseous wastes, and (3) the relative economics and safety of early solidification and immobilization of high-level aqueous wastes as compared with interim storage of liquid wastes to permit the decay of short-lived nuclides.

5.10 Off-Gas Treatment - Task 10

The objective of Task 10 is to develop treatment methods and combinations of methods (systems) for the efficient decontamination of off-gases. The removal of iodine is of primary importance; a lower priority is assigned for the removal of noble gases, tritium, or radioactive particles. Initial emphasis should be placed on studies of the

chemistry of iodine since sufficient knowledge is not available under the various conditions encountered in a fuel processing system to design an off-gas system for processing LMFBR fuels. The chemical studies should be supplemented by the engineering development of methods and systems. Engineering-scale experiments with highly radioactive short-cooled fuels, that is, high iodine concentrations, are required to demonstrate the extremely high decontamination and reliability required for processing LMFBR fuels.

Off-gas processing is directly related to all of the other tasks since the gases from all parts of the plant must pass through the decontamination system prior to discharge. Thus, the species, physical form, and concentration of iodine, as well as the impurity content, must be considered in each processing step. The off-gas treatment system has been divided into three systems: the voloxidizer off-gas (Fig. 5.28), the dissolver off-gas (Fig. 5.29), and the secondary treatment (Fig. 5.30). An option diagram (Fig. 5.31) and a logic diagram (Fig. 5.32) have been prepared which illustrate possible processing methods and systems based on present knowledge and indicate the diverse studies required. A proposed schedule of work is given in Fig. 5.33.

Voloxidizer Off-Gas (Fig. 5.28). - The voloxidizer could receive its carrier gas from the shear. A set of micrometallic blowback filters could be used to retain any PuO_2-UO_2 dust. Tritiated water could possibly be retained by molecular sieves, silica gel, or other inorganic desiccant. The mass of iodine could be retained by either (1) a cold trap or (2) a caustic scrubber. If the voloxidizer is successful, this will represent the bulk of the iodine, and it will be necessary, after decay of the ^{131}I component, to fix and permanently contain the material due to the ^{129}I content. For example, in a 5-ton/day plant processing 30-day-cooled material, the ^{131}I inventory would be 1×10^7 Ci and ^{129}I would accumulate at a rate of 0.26 Ci/day. Fixation methods must be surveyed, and the most attractive methods must be studied (see Sect. 4.9). The gases could pass through one of three alternative units for the removal of noble gases: (1) cryogenic distillation, (2) liquid Freon

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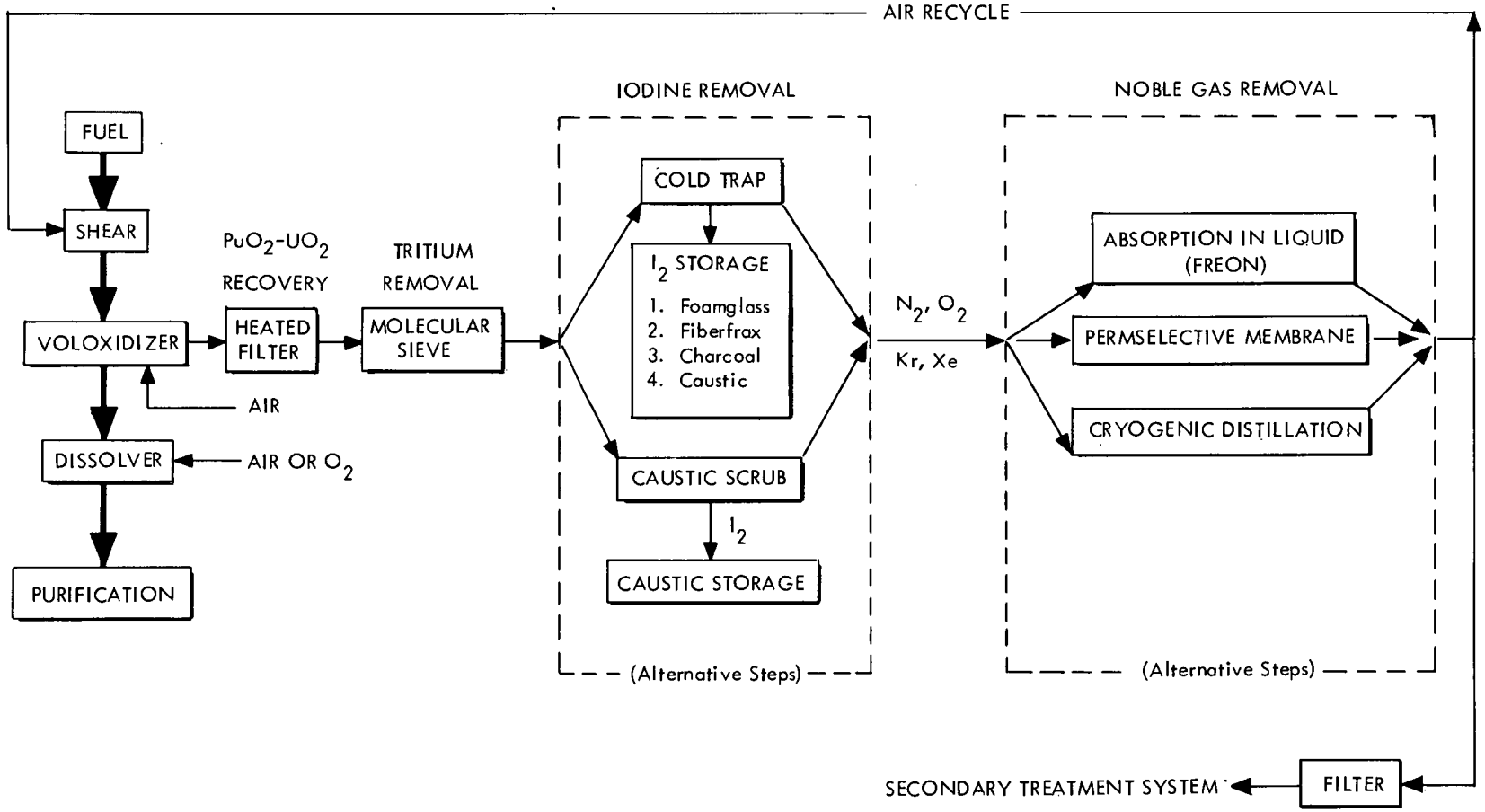

Fig. 5.28. Voloxidizer Off-Gas Treatment System.

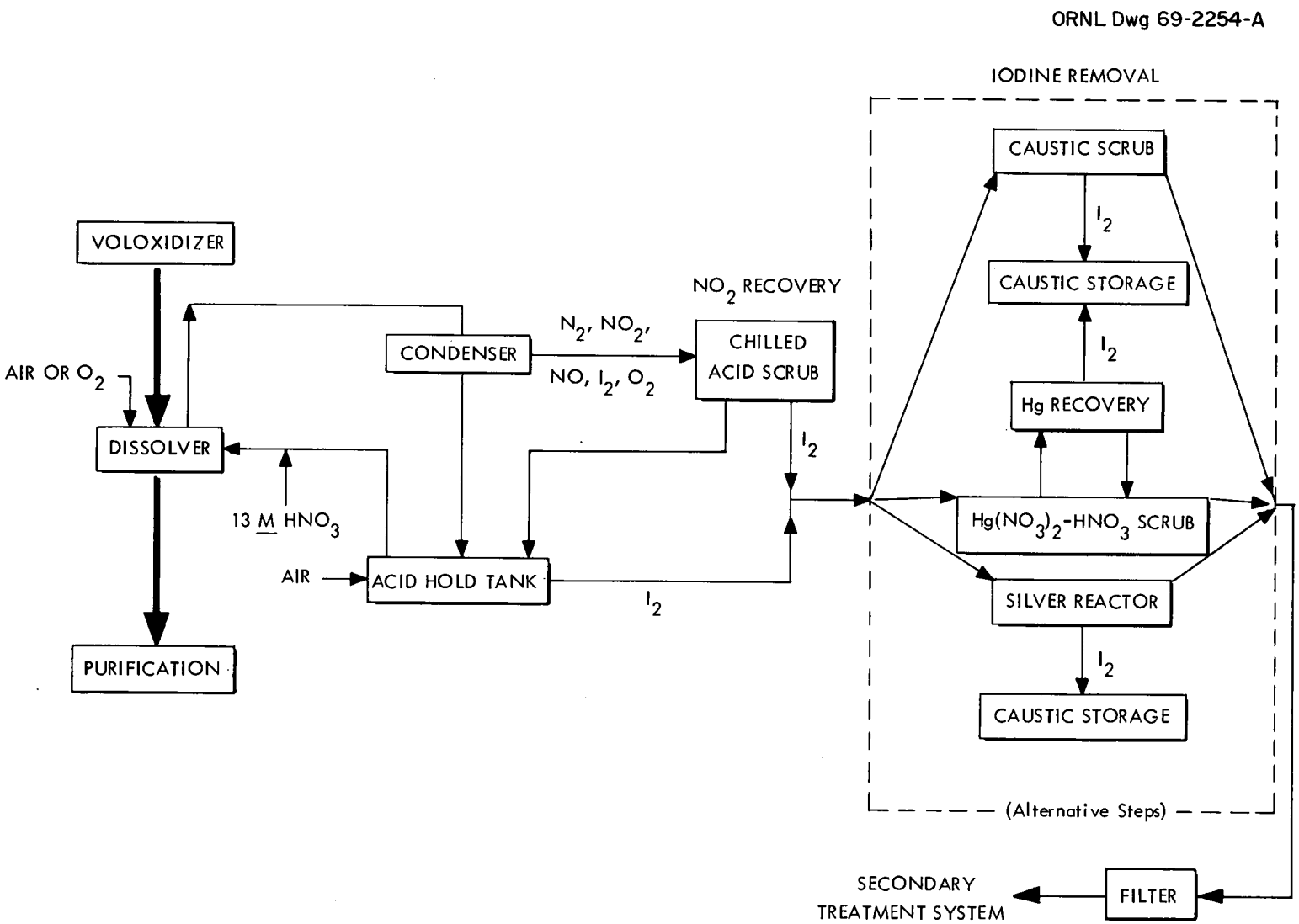

Fig. 5.29. Dissolver Off-Gas Treatment System.

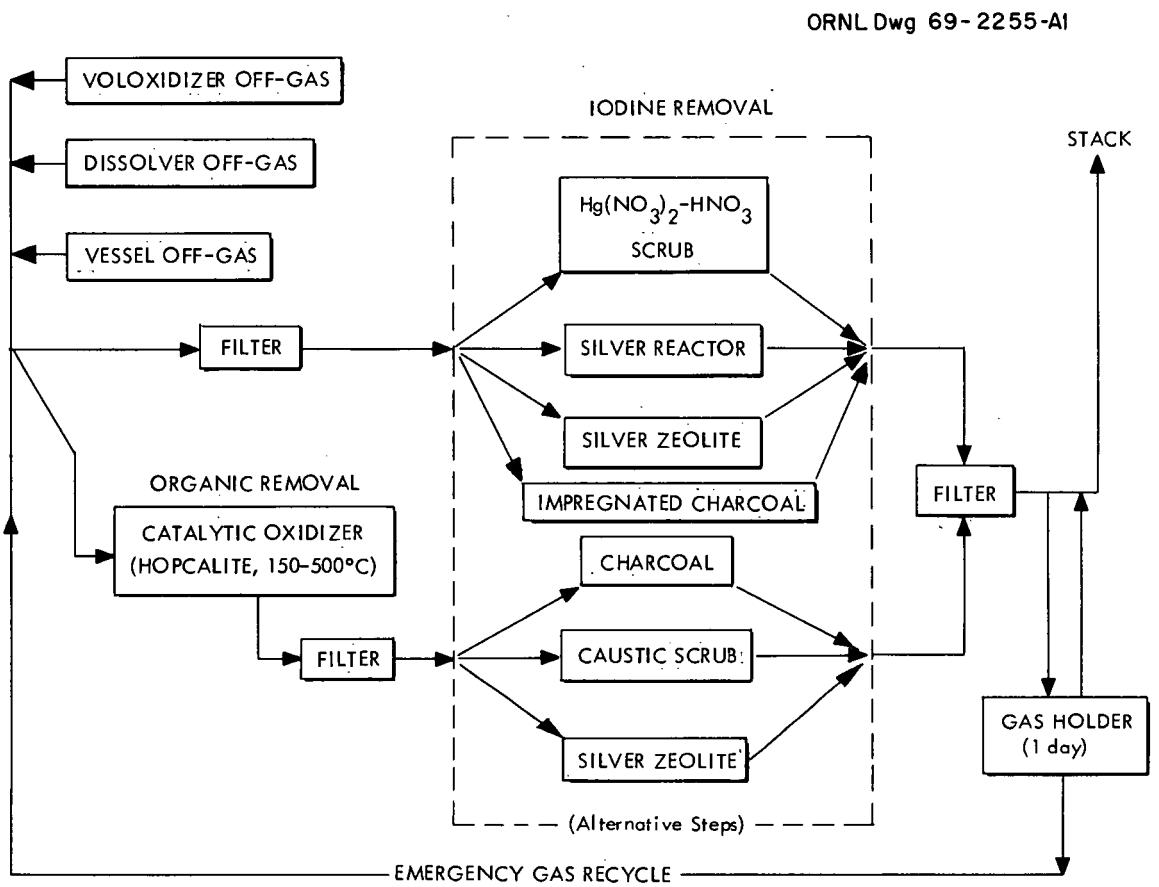

Fig. 5.30. Secondary Treatment System.

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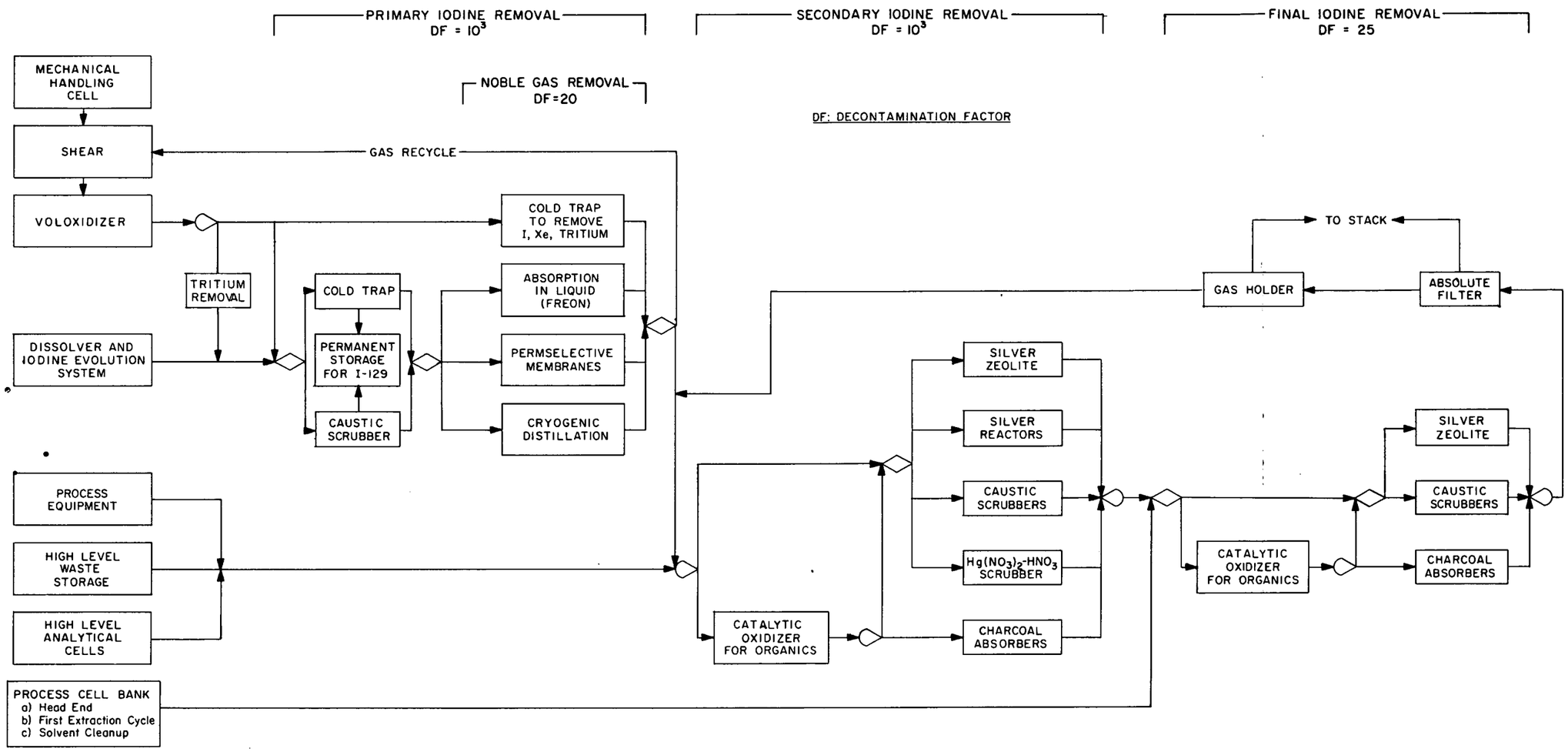

Fig. 5.31. Process Options for Task 10.

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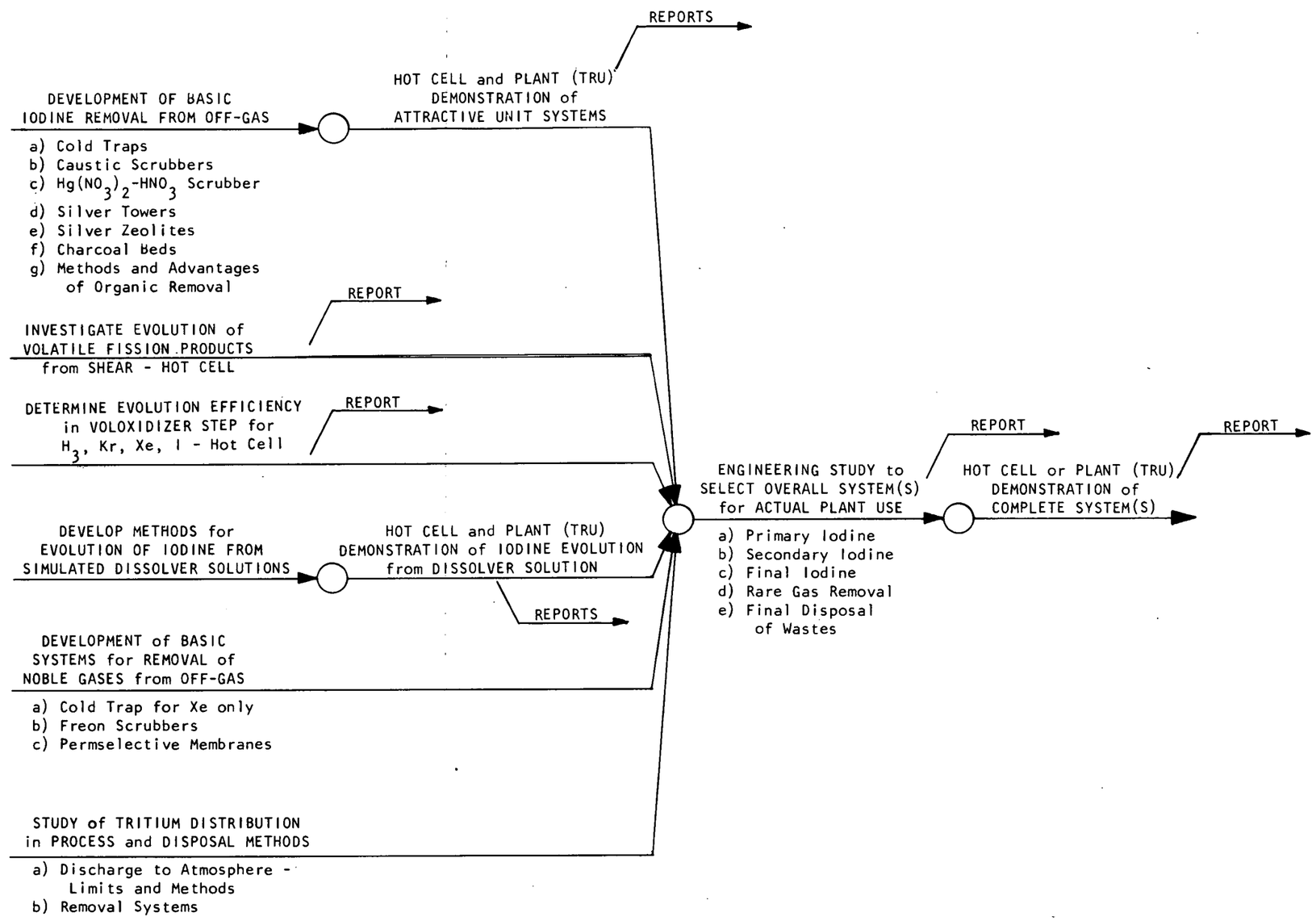

Fig. 5.32. Logic Diagram for Task 10.

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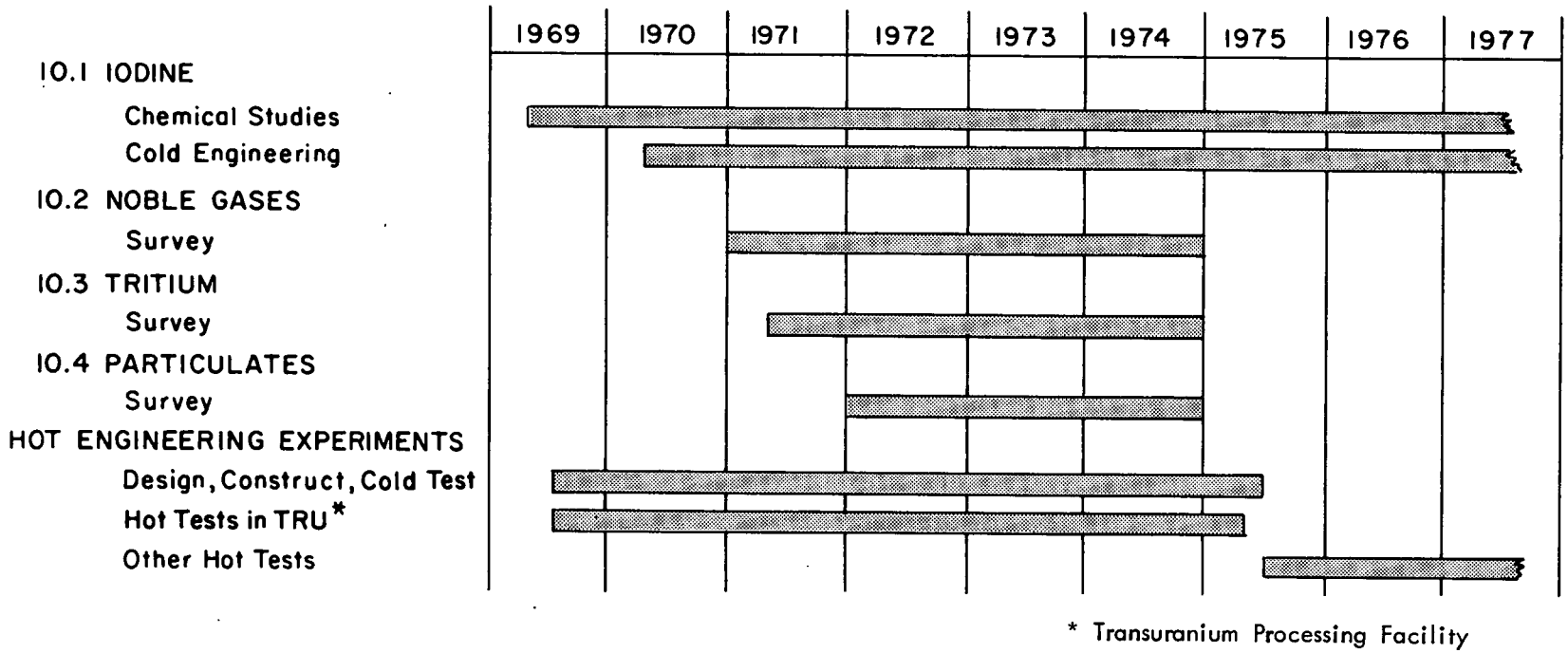

Fig. 5.33. Schedule of Activities for Task 10.

adsorption, or (3) permselective membrane. The above steps would also assist in particle removal.

Dissolver Off-Gas (Fig. 5.29). - Any remaining iodine and noble gases would be volatilized from the dissolver, and most of the iodine would be retained in a hold tank following a downdraft condenser. The iodine may be removed from this acid by sparging prior to its reuse. The noncondensable gases would pass to one or two acid scrubbers for recovery of nitrogen oxides. The gases from the hold tank and the acid scrubber would pass into a caustic or mercuric nitrate--nitric acid scrubber or a silver reactor for removal of the iodine.

Vessel Off-Gas. - The vessel off-gas could pass directly into a secondary system.

Secondary Off-Gas (Fig. 5.30). - The secondary off-gas treatment system would be designed for use as a polishing step after the mass of iodine has been removed. The use of silver-bearing treatment units may be economical if large amounts of iodine are not present. Charcoal filters could be used with safety if the nitrogen oxides have been removed. The treatment steps must be designed to remove iodine from a gas stream that will contain water vapor, organic iodides, elemental iodine, organic compounds, and organic and inorganic particles with adsorbed iodine. As an alternative, the organic vapors and solids could be oxidized to CO_2 and H_2O, and the iodides could be converted to elemental iodine, which can be efficiently removed in subsequent steps. High-temperature oxidation may also vaporize the iodine from inorganic particles and deactivate the surface of the particles. Filtration prior to sorption is desirable because particles tend to pass through sorbent beds and slowly release iodine when retained on a final filter. A silver reactor could also be considered. The hot gases from the oxidizer would be cooled only to the 180°C required for the silver reactor. The hot gases might be treated directly on silver zeolites.

The alternative to elimination (oxidation) of organic material is to treat the gas by methods that can remove iodine efficiently from organic compounds. These methods involve the use of (1) mercuric nitrate--

nitric acid scrubbers, (2) silver zeolites, (3) silver reactors, and (4) impregnated charcoals.

Task 10.1 Iodine

This subtask involves basic chemical and chemical engineering studies of treatment methods and systems for removing iodine from off-gases. It includes: (1) definition of species and concentrations of iodine produced in all plant operations, (2) definition of efficiencies of removal methods and systems (scrubbers, solid sorbents, catalytic oxidizers, and filters) in terms of iodine species, concentration, physical form (particles or gas), chemical form (organic or inorganic compound), and impurities (organic or inorganic vapors, particulates, and nitrogen oxides), and (3) definition of the effect of radiation on these systems. The efficient recovery of nitrogen oxides for reuse as nitric acid in the dissolver system must be considered also.

Initial emphasis will be placed on (1) scrubbers [NaOH, $Hg(NO_3)_2$, HNO_3] and cold trap systems because of their primary role in removing and retaining the mass of iodine, and (2) on catalytic oxidation, which may improve removal efficiencies by converting iodine to a single species. Other promising alternatives will be studied on an appropriate schedule. Chemical studies will be followed by cold engineering research and development to define efficiencies and scale-up factors. An early objective is to design efficient scrubbers that can effectively utilize the high partition coefficients predicted by chemical studies. Hot experiments will be conducted in hot cells and in the Transuranium Processing Plant when appropriate. Hot engineering-scale tests will also be made in conjunction with other engineering tests in the LMFBR development program.

Task 10.2 Noble Gases

After the off-gas system for iodine removal is reasonably well defined, a survey will be made to determine what additional work will be necessary for noble gas removal. The factors affecting the choice of noble gas removal method will be defined in terms of interrelated effects on the iodine removal system, for example, the degree of impurity removal (nitrogen oxides, moisture, etc.) required prior to noble gas removal.

Task 10.3 Tritium

Tritium removal will receive the lowest priority. After the iodine system is reasonably well defined, a survey will be made to determine the best place to remove tritium, and to define any additional development work needed.

Task 10.4 Particulates

A survey of particulate removal methods used in the nuclear industry or being developed in AEC programs will be made to choose units for use in hot engineering experiments. It is expected that adequate removal methods will be available and that additional development is not required other than that associated with Subtasks 10.1, 10.2, and 10.3.

5.11 Radiation and Shielding - Task 11

This task has the objective of developing data on the radiation properties and shielding requirements of LMFBR fuels. Such data are needed in order to promote safe and economical shipment and recovery of these fuels. The impetus for this work should be provided by those who have responsibility for the design and development of processes to be used in the shipping and processing of LMFBR fuel. The work should be coordinated with the related development programs in support of LMFBR fuel fabrication and reactor design.

An idealized logic diagram for the development of radiation control techniques and shields for use in shipping and processing LMFBR fuel is shown in Fig. 5.34. Actually, the paths shown in the diagram should be followed with progressive refinement during the various stages of development that will culminate in conceptual designs, preliminary designs, and detailed designs.

Task 11.1 Radiation Properties of Radionuclides

The objective of this task is to develop the basic nuclear data that are needed to determine the radiation properties and production of individual radionuclides present in spent LMFBR fuel. The end product of this

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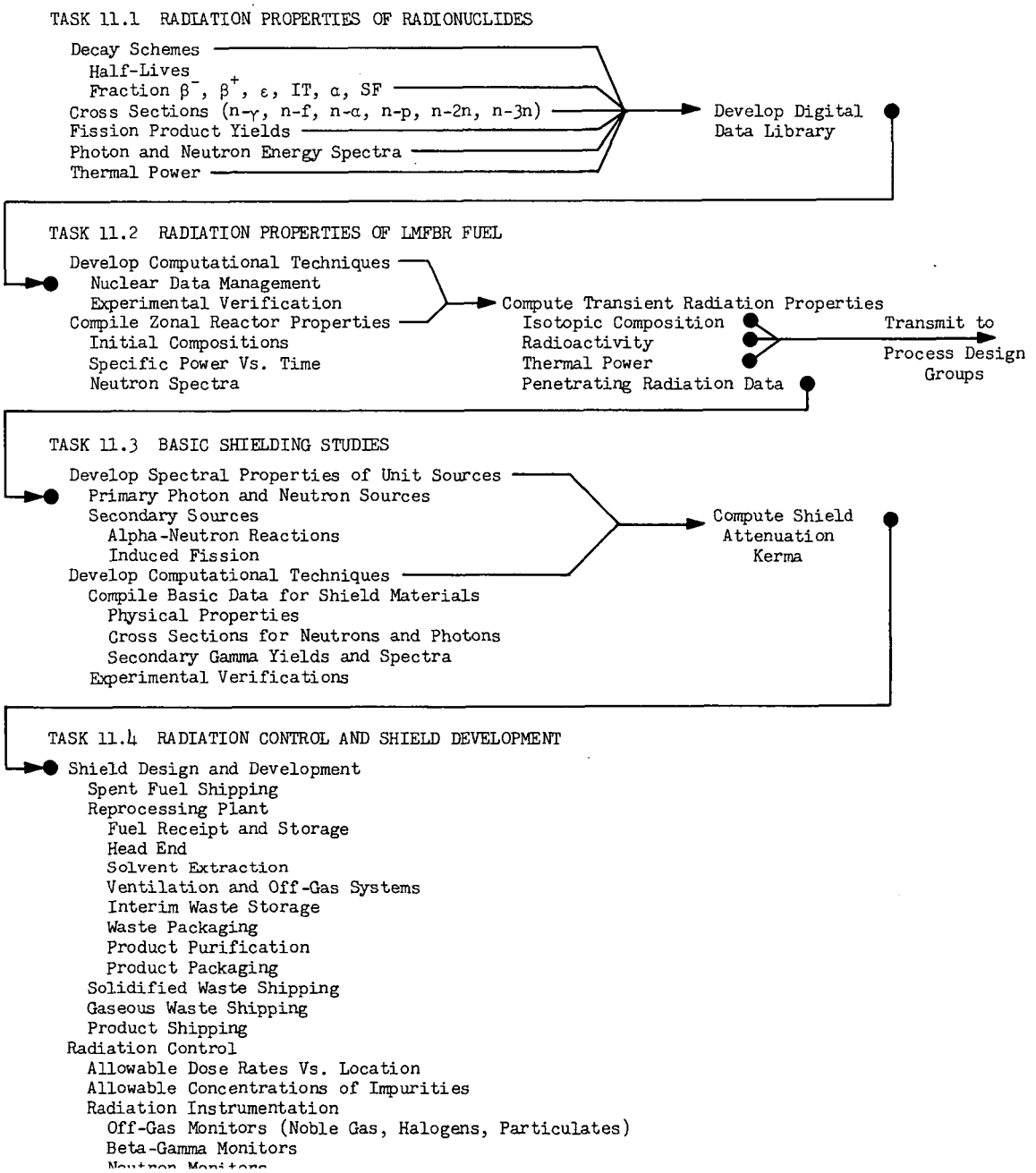

Fig. 5.34. Logic Diagram for Task 11.

task should be an evaluated compilation of nuclear data in a form suitable for use in computing the gross radiation properties of LMFBR fuels.

The data that are needed include decay schemes, cross sections as a function of neutron energy, spectrum-averaged yields of nuclides resulting from the fission of important actinides, energy spectra of photon and neutron radiations, and the thermal power of radioactive disintegrations. Accurate and detailed experimental data should be developed for about 100 nuclides that constitute important sources of penetrating radiation, thermal power, and physiological hazard in LMFBR fuel. Selected experimental data, supplemented by theoretical predictions and correlations of nuclear systematics, are required for approximately 1000 additional nuclides.

Task 11.2 Radiation Properties of LMFBR Fuels

The objective of this task is to determine the radiation properties of LMFBR fuel for studies of chemical separations processes, radiation damage, heat dissipation requirements, containment requirements, and shielding. Computer codes, such as ORIGEN, should be modified and refined to permit incorporation of a more detailed description of the reactor neutron spectrum as a function of position and time, and photon and neutron spectra of primary radiations from irradiated fuel. The computational methods should be tested and refined by a thorough analysis of fuel specimens that have been irradiated in LMFBR demonstration and fast test reactors.

Task 11.3 Basic Shielding Studies

The objective of this task is to develop data on shields and on shielding combinations that will be required in the shipment and recovery of LMFBR fuels. The work requires development of the spectral distribution of photons and neutrons in the various process operations (spent fuel shipping, off-gas cleanup systems, waste streams, plutonium purification system, etc.), developing computational procedures, and determining dose attenuation kerma for shielding materials of principal interest. The computational methods should be validated and refined by experimental

shielding studies using sources consisting of fission products, highly burned plutonium, and transplutonium isotopes.

Task 11.4 Radiation Control and Shield Development

The purpose of this task is to design and develop shields and to formulate radiation control techniques that are needed for the various operations in the shipment and recovery of LMFBR fuels. The shielding studies should consider the requirements for direct access, viewing, structural components, and shield penetrations. The radiation control studies should develop allowable exposure rates in various operating areas, determine specifications for sources of radiation in product materials, and develop instrumentation for monitoring radiation levels in gaseous effluents and in operating areas.

5.12 Criticality - Task 12

This task has the objective of developing criticality data and criticality control techniques that are needed for the safe and economical shipment and recovery of LMFBR fuels. The work requires coordination between groups in private industry and the National Laboratories that are involved in (1) process design and development, (2) development of computational methods, and (3) experimentation to generate basic nuclear data, criticality data, and techniques for the assay of fissile materials. The impetus for the criticality studies of each process operation (shipment, dissolution, solvent extraction, etc.) should be provided by those who have responsibility for designing and evaluating the operational procedure; a summary of the logical development of criticality data for a given process operation is presented in Fig. 5.35. In addition, provision should be made for studies that are divorced from specific process operations but have the purpose of developing a unified and comprehensive understanding of the criticality parameters of systems containing plutonium and plutonium-uranium mixtures.

This task is divided into seven subtasks. Tasks 12.1 through 12.4 have the purpose of developing criticality data for the nuclear systems

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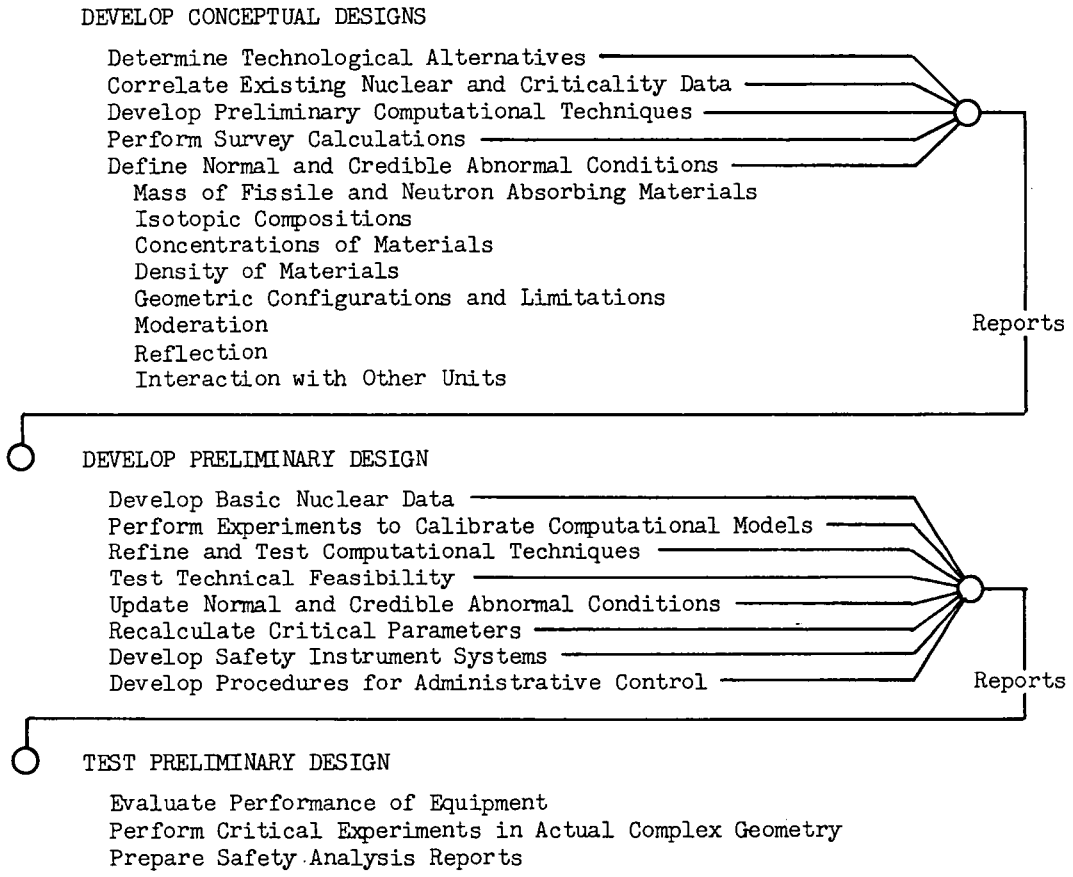

Fig. 5.35. Logic Diagram for Nuclear Safety Development of Each Process Operation.

that are expected to be encountered in the shipment and aqueous processing of spent LMFBR fuels. A summary of the major requirements for criticality data in the various process operations is presented in Table 5.1. The uranium-plutonium compositions of principal interest in these studies are presented in Table 5.2. Other process parameters are presented elsewhere in this report. Tasks 12.5 and 12.6 have the objectives of developing computational techniques and generating comprehensive correlations of criticality data. Task 12.7 has the objective of developing nondestructive techniques for accountability and control of fissile materials.

Task 12.1 Arrays of LMFBR Elements

This task has the purpose of developing criticality data for application to the shipment and storage of LMFBR elements. Arrays of core--axial blanket elements are of primary concern. The criticality limitations for safe arrays of such elements should assume the presence of elements that will introduce maximum reactivity into the array, namely, elements that are normally unirradiated and are from the outer portion of a core zone. This analysis should also consider credible configurations of damaged elements. The determinations of the effective neutron multiplication factor as a function of the size and the arrangement of the arrays (both of which are applicable primarily to the determination of the neutron source strength in shipping) should assume credible arrays of elements in their irradiated condition.

Task 12.2 UO_2-PuO_2 Solids at Low Moderation

This task has the purpose of determining the criticality parameters of dry chopped elements, UO_2-PuO_2 solids, and PuO_2 solids. The moderator would be (1) the gas in a shear or a voloxidizer, (2) the molten stainless steel in a melt-decladding step, or (3) other molten metals in possible metallurgical decladding steps. This task should also include the development of data for arrays of subcritical units of PuO_2; these data will permit the design of product storage arrays and product shipping containers.

Table 5.1. Summary of Major Requirements for Criticality Data in the Aqueous Processing of Spent LMFBR UO_2-PuO_2 Fuels

Process Operation	Nuclear System: Fuel(s)	Nuclear System: Moderator(s)	Nuclear System: Reflector(s)	Nuclear Parameters to Determine
Spent Fuel Shipping	Arrays of core fuel elements	1. Sodium 2. Lithium 3. Other liquid metal 4. Air	1. Thick iron 2. Thick lead	K_{eff} vs size of array considering: Spacing of elements Interspersal of radial blanket elements Neutron absorbing structural materials Moderator Reflector Reactor type (fresh and discharged composition)
Fuel Element Storage	Arrays of core fuel elements	1. Water 2. Sodium 3. Air (gas)	Infinite system	Critical spacing of elements vs configuration of array and type of moderator
Head End	Chopped elements	1. Air (gas) 2. Molten stainless steel 3. Molten zinc	1 in. of iron plus insulation	Critical mass, volume, and dimensions vs fuel density and type of moderator
Dissolver	1. UO_2-PuO_2 solids in fissile solution 2. Above with fixed poison basket 3. PuO_2 solids in fissile solution	1. U-Pu nitrate solution 2. Solution as in No. 1, but containing B or Cd in fissile solution	1. 1 in. of iron 2. Thick water 3. Thick concrete 4. Solution	Critical mass, volume, and dimensions for credible combinations of fuel, moderator, and reflector
Feed Adjustment	1. Homogeneous U-Pu nitrate solution 2. Solution as in No. 1, but containing B or Cd	Fissile solution	1. 1 in. of iron 2. Infinite system	Critical mass, volume, and dimensions vs moderating ratio Limiting critical concentrations of U-Pu solutions
Solvent Extraction[a]	1. Homogeneous aqueous or organic solution of Pu or U-Pu 2. Solution as in No. 1, but in fixed poison array	Solution or ion exchange resin	1. 1 in. of iron 2. Thick water	Critical dimensions vs moderating ratio and type of fixed absorber array Limiting critical concentrations of Pu solutions
Plutonium Solution Storage	1. Homogeneous Pu nitrate solution 2. Above in tanks packed with borosilicate glass	Solution	1. 1 in. of iron 2. Thick water	Critical dimensions of units in a closely spaced array Safe concentration in infinite system packed with glass
Plutonium Oxide Storage	Arrays of packaged PuO_2	1. Air 2. Water 3. Plastic	1/2 in. of steel	Mass and dimensions of units Critical number of units in array

[a]The requirements for this operation are also applicable to ion exchange and product evaporation.

Table 5.2. Uranium-Plutonium Compositions of Principal Interest in LMFBR Criticality Studies

Type of Fuel	Composition (Weight Fraction)						
	$\frac{Pu}{(U + Pu)}$	$\frac{^{235}U}{U}$	$\frac{^{238}Pu}{Pu}$	$\frac{^{239}Pu}{Pu}$	$\frac{^{240}Pu}{Pu}$	$\frac{^{241}Pu}{Pu}$	$\frac{^{242}Pu}{Pu}$
Fast Test Reactor							
Charge	0.20-0.25	0.007	0.000	0.895	0.100	0.005	0.000
Discharge	0.20-0.24	0,007	0.00	0.85	0.14	0.01	0.00
Typical LMFBR							
Charge to core	0.13-0.22	0.002	0.01	0.59	0.24	0.12	0.04
Discharge							
Core	0.12-0.21	0.002	0.01	0.60	0.27	0.07	0.05
Core + axial blanket	0.09-0.14	0.002	0.01	0.63	0.25	0.06	0.05
Radial blanket	0.01-0.03	0.002	0.00	0.98	0.02	0.00	0.00

Task 12.3 UO_2-PuO_2 Solids in Fissile Solution

This task has the purpose of developing criticality parameters for the mixtures of fissile solids and the fissile solution that may result from the dissolution of spent LMFBR fuel. The solids are assumed to be either UO_2-PuO_2 or pure PuO_2; the latter could result if PuO_2 were precipitated or if the UO_2 were dissolved selectively. The fissile solution is assumed to be an aqueous uranium-plutonium nitrate solution, probably containing boron or cadmium as a soluble neutron absorber. In addition, some dissolver concepts will include arrays of fixed neutron absorbers such as stainless steel containing boron or gadolinium.

Task 12.4 Homogeneous Solutions of Uranium-Plutonium

This task has the objective of developing criticality parameters for homogeneous aqueous or organic solutions of plutonium or uranium-plutonium. Limiting critical concentrations of aqueous solutions of uranium-plutonium, with and without soluble neutron absorbers, must be known for analysis of the feed adjustment operation. Knowledge of critical dimensions and of the effects of arrays of fixed neutron absorbers is needed for the design of solvent extraction, ion exchange, and product evaporation systems. Critical dimensions for the plutonium nitrate solutions, as well as critical concentrations of these solutions in large tanks packed with borosilicate-glass raschig rings, are needed for the evaluation of plutonium solution storage systems.

Task 12.5 Computational Techniques

The objective of this task is to develop computational techniques that may be used for reliable and accurate predictions of criticality parameters for systems containing uranium and plutonium. This work includes compilation and evaluation of basic nuclear data and testing and refinement of the computational techniques as experimental criticality data become available. The task requires up-to-date information with respect to the types of systems for which calculations are required and timely communications with experimental groups who will develop the basic nuclear and criticality data.

Task 12.6 Evaluated Criticality Data Compilation

The purpose of this task is to compile criticality data for uranium-plutonium systems using the widest range of variables (i.e., plutonium concentration, isotopic compositions, moderation, reflection, interaction, type of chemical compound, etc.) possible by the theoretical analysis of experimental data and the application of reliable computational techniques. The objective is to provide a compilation that is not related to specific process operations or to specific types of fuels, but one that can constitute a basis for exercising future options.

Task 12.7 Nondestructive Assay of Fissile Materials

This task has the objective of developing nondestructive techniques for determining the fissile content of highly radioactive fissile materials. The work on the assay of unirradiated fissile materials, which has been funded primarily through the USAEC Office of Nuclear Materials Safeguards, should be extended to systems containing intense sources of neutron and gamma radiation. The needed techniques include those for: (1) assaying spent fuel elements, (2) assaying solid wastes from a fuel processing plant, and (3) monitoring process vessels for subcriticality.

A reliable nondestructive technique for assaying spent fuel elements would increase the flexibility of the fuel processor by minimizing the requirement for equipment cleanout between campaigns. Since the technique could permit completion of the fuel transfer contract before actual recovery of the fuel, the processor could recover fuels of the most advantageous type, regardless of their order of receipt or postirradiation decay time. The technical and economic feasibility of alternative techniques for the nondestructive assay of spent LMFBR fuel should be studied. The level of effort in experimentation, development, and design should be based on the potential value of the technique and the probability of success.

Methods should be developed for the assay of solid wastes from fuel processing plants. A complete material balance and the prevention of

fissile accumulations in fuel processing plants require the assay of such wastes as chopped hulls, metal ingots, and degraded ion exchange resins. Techniques that are being developed for use in plants for processing fuels from light-water reactors should be modified and refined for application to plants for processing LMFBR fuels.

Monitors are needed to signal the accumulation of fissile solids in process vessels. This need will be especially acute in LMFBR fuel processing because of the high plutonium content of the fuel and the possibility of plutonium polymerization. Process vessels that probably will require monitoring (because of the accumulation of fissile solids as undissolved fuel, cruds, or precipitates) include dissolvers, feed adjustment tanks, solvent extraction equipment, and evaporators. Neutron interrogation techniques should be investigated as possible bases for such monitors.

5.13 Engineering Studies - Task 13

Certain engineering studies are needed in addition to the studies that will be performed under the tasks previously enumerated. These studies should be concerned with general plant problems, and their results should be used to guide the overall development effort. In addition, conceptual studies relating to interim fuel processing as well as to long-term processing should be made. The subjects of the proposed studies are as follows:

1. An in-depth review of available facilities for processing of FFTF and early demonstration reactor fuels should be made. This review should determine what minimum modifications will be required to allow the processing of long-decayed fuel. The areas of uncertainty requiring additional development effort should be pointed out. This study should include the means for shipping and receiving these fuels.

2. The results of a criticality assessment (see Task 12) of the various unit operations and processes incidental to LMFBR fuel

recovery should be studied for their overall impact on plant design. Study of these results will also permit us to determine whether a significant relaxation in size or concentration limits might be available through extension of current knowledge of physical and chemical behavior.

3. A study should be made of the available information concerning plutonium polymerization to indicate what changes in equipment design or operational procedures will be required to avoid this problem.

4. A study to indicate the relationship of overall fuel cycle cost to decay time should be performed. The primary effort in this area should be the determination of the variation in plant cost with the addition of provisions for gaseous fission product retention and of special provisions for handling fuels having high rates of decay heat generation.

5. A study should be made to determine the containment requirements for radioactive materials in LMFBR fuel recovery plants. This study should consider such possible mechanisms for release of radioactive materials as fires, explosions, and nuclear excursions. Safe shutdown capability should be provided for credible external events such as earthquakes and tornados.

6. A study should be made to determine the impact on overall investment and processing cost of variations in the nature of equipment used for recovering fissile and fertile materials from spent LMFBR fuels. This study should include an evaluation of the merits of continuous equipment for certain operations that have traditionally been batch operations (for instance, the dissolving operation) as well as the effect of equipment physical size and other characteristics on overall plant cost.

7. A study of the requirements for source and fissile material accountability is needed. This investigation will require consultations with various prospective fuel owners and current

fuel processors, as well as with government regulatory authorities, concerning the means that are currently used and that might possibly be used in this type of work.

8. A study should be made to ascertain the most acceptable method for recovering fuel from early commercial reactors. This method should also be the most suitable one for use during the period when a fast breeder reactor economy is developing. The purpose of this study should be to determine how the fuel load can best be processed while minimizing capital investment, but yet establishing a technology for processing after reduced decay periods with rapid throughput and with the prospect for costs that are competitive with LWR processing.

9. An outline type of conceptual design should be prepared for a facility to be used for recovering large amounts of fuel from a mature fast breeder reactor complex. This conceptual design should, in effect, integrate the results of the work performed under the various other tasks and, in turn, indicate the magnitude of the effort required to provide facilities for the large-scale recovery of spent LMFBR fuels. This design should provide an estimate of the variation in operational cost as a function of decay time and throughput.

10. A study should be initiated to determine the economic viability of an on-site processing facility serving a single "power park" that generates approximately 15,000 Mw(electrical).

APPENDIX A: CONCENTRATIONS, RADIOACTIVITY, AND THERMAL POWER OF FISSION PRODUCTS IN THE CORE, AXIAL BLANKET, AND RADIAL BLANKET OF THE ATOMICS INTERNATIONAL REFERENCE OXIDE LMFBR

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
KR 83	0.0	1.61E 02	1.61E 02	1.61E 02	1.61E 02	1.61E 02	1.61E 02
KR 84	0.0	2.78E 02	2.78E 02	2.78E 02	2.78E 02	2.78E 02	2.78E 02
KR 85	0.0	6.35E 01	6.31E 01	6.25E 01	6.18E 01	5.23E 01	9.19E 00
RB 85	0.0	1.03E 02	1.03E 02	1.04E 02	1.05E 02	1.14E 02	1.57E 02
KR 86	0.0	2.49E 02	2.49E 02	2.49E 02	2.49E 02	2.49E 02	2.49E 02
RB 87	0.0	3.07E 02	3.07E 02	3.07E 02	3.07E 02	3.07E 02	3.07E 02
SR 88	0.0	4.73E 02	4.73E 02	4.73E 02	4.73E 02	4.73E 02	4.73E 02
SR 89	0.0	7.92E 01	5.31E 01	2.39E 01	1.07E 01	3.58E-05	0.0
Y 89	0.0	4.92E 02	5.19E 02	5.48E 02	5.61E 02	5.72E 02	5.72E 02
SR 90	0.0	7.28E 02	7.27E 02	7.24E 02	7.21E 02	6.76E 02	3.47E 02
ZR 90	0.0	1.37E 01	1.52E 01	1.82E 01	2.11E 01	6.57E 01	3.95E 02
Y 91	0.0	1.26E 02	8.88E 01	4.38E 01	2.16E 01	3.11E-04	0.0
ZR 91	0.0	6.80E 02	7.17E 02	7.63E 02	7.85E 02	8.06E 02	8.06E 02
ZR 92	0.0	1.06E 03	1.06E 03	1.06E 03	1.06E 03	1.06E 03	1.06E 03
ZR 93	0.0	1.33E 03	1.33E 03	1.33E 03	1.33E 03	1.33E 03	1.33E 03
ZR 94	0.0	1.51E 03	1.51E 03	1.51E 03	1.51E 03	1.51E 03	1.51E 03
ZR 95	0.0	3.33E 02	2.42E 02	1.27E 02	6.72E 01	2.81E-03	0.0
NB 95	0.0	1.77E 02	1.66E 02	1.13E 02	6.77E 01	3.27E-03	0.0
MO 95	0.0	1.39E 03	1.50E 03	1.66E 03	1.77E 03	1.90E 03	1.90E 03
ZR 96	0.0	1.77E 03	1.77E 03	1.77E 03	1.77E 03	1.77E 03	1.77E 03
MO 96	0.0	3.39E 01	3.40E 01	3.40E 01	3.40E 01	3.40E 01	3.40E 01
MO 97	0.0	2.41E 03	2.41E 03	2.41E 03	2.41E 03	2.41E 03	2.41E 03
MO 98	0.0	2.07E 03	2.07E 03	2.07E 03	2.07E 03	2.07E 03	2.07E 03
TC 99	0.0	2.09E 03	2.10E 03	2.10E 03	2.10E 03	2.10E 03	2.10E 03
MO100	0.0	2.46E 03	2.46E 03	2.46E 03	2.46E 03	2.46E 03	2.46E 03
RU100	0.0	5.31E 01	5.31E 01	5.31E 01	5.31E 01	5.31E 01	5.31E 01
RU101	0.0	2.00E 03	2.00E 03	2.00E 03	2.00E 03	2.00E 03	2.00E 03
RU102	0.0	2.26E 03	2.26E 03	2.26E 03	2.26E 03	2.26E 03	2.26E 03
RU103	0.0	2.24E 02	1.32E 02	4.63E 01	1.62E 01	1.04E-06	0.0
RH103	0.0	1.88E 03	1.97E 03	2.05E 03	2.08E 03	2.10E 03	2.10E 03
RU104	0.0	2.13E 03	2.13E 03	2.13E 03	2.13E 03	2.13E 03	2.13E 03
PD105	0.0	1.36E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03
RU106	0.0	1.02E 03	9.63E 02	8.60E 02	7.63E 02	1.29E 02	1.05E-06
PD106	0.0	6.75E 02	7.31E 02	8.35E 02	9.27E 02	1.57E 03	1.69E 03
PD107	0.0	1.24E 03	1.24E 03	1.24E 03	1.24E 03	1.24E 03	1.24E 03
PD108	0.0	7.96E 02	7.96E 02	7.96E 02	7.96E 02	7.96E 02	7.96E 02
AG109	0.0	9.36E 02	9.38E 02	9.38E 02	9.38E 02	9.38E 02	9.38E 02
CD110	0.0	4.97E 01	4.97E 01	4.99E 01	5.00E 01	5.06E 01	5.07E 01
CD111	0.0	1.54E 02	1.57E 02	1.58E 02	1.58E 02	1.58E 02	1.58E 02
CD112	0.0	9.65E 01	9.68E 01	9.68E 01	9.68E 01	9.68E 01	9.68E 01
SB121	0.0	3.08E 01	3.09E 01	3.09E 01	3.09E 01	3.10E 01	3.18E 01
SB125	0.0	4.78E 01	4.80E 01	4.61E 01	4.42E 01	2.28E 01	2.22E-02
SN126	0.0	1.55E 02	1.55E 02	1.55E 02	1.55E 02	1.55E 02	1.55E 02
TE127M	0.0	1.93E 01	1.66E 01	1.13E 01	7.72E 00	1.88E-02	0.0
I127	0.0	2.85E 02	2.92E 02	2.97E 02	3.00E 02	3.08E 02	3.08E 02
TE128	0.0	4.07E 02	4.08E 02	4.08E 02	4.08E 02	4.08E 02	4.08E 02
TE129M	0.0	2.74E 01	1.50E 01	4.40E 00	1.30E 00	5.46E-09	0.0

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
I129	0.0	7.96E 02	8.08E 02	8.19E 02	8.22E 02	8.23E 02	8.23E 02
TE130	0.0	8.84E 02	8.84E 02	8.84E 02	8.84E 02	8.84E 02	8.84E 02
XE130	0.0	2.58E 01	2.58E 01	2.58E 01	2.58E 01	2.58E 01	2.58E 01
XE131	0.0	1.62E 03	1.66E 03	1.66E 03	1.66E 03	1.66E 03	1.66E 03
XE132	0.0	2.36E 03	2.38E 03	2.38E 03	2.38E 03	2.38E 03	2.38E 03
CS133	0.0	2.99E 03	3.04E 03	3.05E 03	3.05E 03	3.05E 03	3.05E 03
XE134	0.0	3.35E 03	3.35E 03	3.35E 03	3.35E 03	3.35E 03	3.35E 03
CS134	0.0	6.04E 01	5.87E 01	5.56E 01	5.26E 01	2.19E 01	2.43E-03
BA134	0.0	1.04E 01	1.20E 01	1.52E 01	1.82E 01	4.89E 01	7.08E 01
CS135	0.0	3.25E 03	3.25E 03	3.25E 03	3.25E 03	3.25E 03	3.25E 03
XE136	0.0	2.96E 03	2.96E 03	2.96E 03	2.96E 03	2.96E 03	2.96E 03
BA136	0.0	9.15E 01	9.55E 01	9.64E 01	9.65E 01	9.65E 01	9.65E 01
CS137	0.0	3.06E 03	3.05E 03	3.04E 03	3.03E 03	2.85E 03	1.53E 03
BA137	0.0	5.23E 01	5.81E 01	6.96E 01	8.12E 01	2.57E 02	1.58E 03
BA138	0.0	2.96E 03	2.96E 03	2.96E 03	2.96E 03	2.96E 03	2.96E 03
LA139	0.0	2.77E 03	2.77E 03	2.77E 03	2.77E 03	2.77E 03	2.77E 03
BA140	0.0	8.72E 01	1.72E 01	6.67E-01	2.59E-02	0.0	0.0
CE140	0.0	2.96E 03	3.04E 03	3.06E 03	3.06E 03	3.06E 03	3.06E 03
CE141	0.0	2.36E 02	1.25E 02	3.46E 01	9.58E 00	1.56E-08	0.0
PR141	0.0	2.49E 03	2.60E 03	2.69E 03	2.72E 03	2.73E 03	2.73E 03
CE142	0.0	2.41E 03	2.41E 03	2.41E 03	2.41E 03	2.41E 03	2.41E 03
ND142	0.0	1.24E 01	1.25E 01	1.25E 01	1.25E 01	1.25E 01	1.25E 01
PR143	0.0	9.57E 01	2.78E 01	1.33E 00	6.41E-02	1.05E-22	0.0
ND143	0.0	2.41E 03	2.50E 03	2.53E 03	2.53E 03	2.53E 03	2.53E 03
CE144	0.0	1.04E 03	9.67E 02	8.36E 02	7.22E 02	7.17E 01	2.53E-09
ND144	0.0	9.94E 02	1.07E 03	1.20E 03	1.31E 03	1.96E 03	2.03E 03
ND145	0.0	1.55E 03	1.55E 03	1.55E 03	1.55E 03	1.55E 03	1.55E 03
ND146	0.0	1.45E 03	1.45E 03	1.45E 03	1.45E 03	1.45E 03	1.45E 03
PM147	0.0	8.75E 02	8.85E 02	8.53E 02	8.16E 02	4.11E 02	3.25E-01
SM147	0.0	1.74E 02	1.93E 02	2.31E 02	2.63E 02	6.73E 02	1.08E 03
ND148	0.0	9.00E 02	9.00E 02	9.00E 02	9.00E 02	9.00E 02	9.00E 02
SM148	0.0	8.36E 01	8.81E 01	9.14E 01	9.27E 01	9.34E 01	9.34E 01
SM149	0.0	6.62E 02	6.67E 02	6.67E 02	6.67E 02	6.67E 02	6.67E 02
ND150	0.0	5.30E 02	5.30E 02	5.30E 02	5.30E 02	5.30E 02	5.30E 02
SM150	0.0	5.36E 01	5.36E 01	5.36E 01	5.36E 01	5.36E 01	5.36E 01
SM151	0.0	4.01E 02	4.02E 02	4.02E 02	4.01E 02	3.93E 02	3.17E 02
SM152	0.0	3.66E 02	3.66E 02	3.66E 02	3.66E 02	3.66E 02	3.66E 02
EU153	0.0	1.77E 02	1.78E 02	1.78E 02	1.78E 02	1.78E 02	1.78E 02
SM154	0.0	1.47E 02	1.47E 02	1.47E 02	1.47E 02	1.47E 02	1.47E 02
EU154	0.0	1.74E 01	1.74E 01	1.73E 01	1.71E 01	1.53E 01	4.75E 00
EU155	0.0	1.64E 02	1.59E 02	1.49E 02	1.40E 02	5.19E 01	1.68E-03
GD155	0.0	4.76E 01	5.26E 01	6.23E 01	7.14E 01	1.59E 02	2.11E 02
GD156	0.0	1.19E 02	1.23E 02	1.24E 02	1.24E 02	1.24E 02	1.24E 02
GD157	0.0	7.04E 01	7.05E 01	7.05E 01	7.05E 01	7.05E 01	7.05E 01
GD158	0.0	1.97E 02	1.97E 02	1.97E 02	1.97E 02	1.97E 02	1.97E 02
TB159	0.0	9.71E 01	9.74E 01	9.74E 01	9.74E 01	9.74E 01	9.74E 01
DY161	0.0	1.96E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01
SUBTOT	0.0	8.43E 04	8.45E 04	8.46E 04	8.46E 04	8.45E 04	8.44E 04
TOTALS	0.0	8.47E 04	8.47E 04	8.47E 04	8.47E 04	8.47E 04	8.47E 04

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	2.42E-01	2.41E-01	2.38E-01	2.36E-01	2.04E-01	4.46E-02
ZN	0.0	1.40E-03	3.05E-08	1.45E-17	0.0	0.0	0.0
GA	0.0	4.26E-04	1.33E-08	6.31E-18	0.0	0.0	0.0
GE	0.0	3.53E-01	3.57E-01	3.57E-01	3.57E-01	3.57E-01	3.57E-01
AS	0.0	2.64E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02
SE	0.0	9.69E 00	9.69E 00	9.69E 00	9.69E 00	9.69E 00	9.69E 00
BR	0.0	3.34E 00	3.28E 00	3.28E 00	3.28E 00	3.28E 00	3.28E 00
KR	0.0	7.51E 02	7.51E 02	7.50E 02	7.50E 02	7.40E 02	6.97E 02
RB	0.0	4.10E 02	4.11E 02	4.11E 02	4.12E 02	4.22E 02	4.65E 02
SR	0.0	1.28E 03	1.25E 03	1.22E 03	1.21E 03	1.15E 03	8.22E 02
Y	0.0	6.21E 02	6.08E 02	5.92E 02	5.83E 02	5.72E 02	5.72E 02
ZR	0.0	6.69E 03	6.64E 03	6.57E 03	6.54E 03	6.54E 03	6.87E 03
NB	0.0	1.78E 02	1.66E 02	1.14E 02	6.78E 01	5.56E-03	1.88E-02
MO	0.0	8.38E 03	8.47E 03	8.64E 03	8.74E 03	8.88E 03	8.88E 03
TC	0.0	2.09E 03	2.10E 03	2.10E 03	2.10E 03	2.10E 03	2.10E 03
RU	0.0	7.69E 03	7.55E 03	7.36E 03	7.24E 03	6.58E 03	6.45E 03
RH	0.0	1.88E 03	1.97E 03	2.05E 03	2.08E 03	2.10E 03	2.10E 03
PD	0.0	4.08E 03	4.14E 03	4.25E 03	4.34E 03	4.98E 03	5.10E 03
AG	0.0	9.41E 02	9.39E 02	9.39E 02	9.39E 02	9.38E 02	9.38E 02
CD	0.0	3.21E 02	3.24E 02	3.25E 02	3.25E 02	3.25E 02	3.24E 02
IN	0.0	4.18E 00	4.22E 00	4.25E 00	4.27E 00	4.45E 00	5.41E 00
SN	0.0	1.85E 02	1.84E 02	1.84E 02	1.84E 02	1.83E 02	1.83E 02
SB	0.0	8.50E 01	8.15E 01	7.96E 01	7.78E 01	5.65E 01	3.45E 01
TE	0.0	1.37E 03	1.34E 03	1.32E 03	1.32E 03	1.33E 03	1.35E 03
I	0.0	1.12E 03	1.10E 03	1.12E 03	1.12E 03	1.13E 03	1.13E 03
XE	0.0	1.04E 04	1.04E 04	1.04E 04	1.04E 04	1.04E 04	1.04E 04
CS	0.0	9.37E 03	9.41E 03	9.40E 03	9.38E 03	9.18E 03	7.83E 03
BA	0.0	3.20E 03	3.14E 03	3.14E 03	3.15E 03	3.36E 03	4.70E 03
LA	0.0	2.78E 03	2.77E 03	2.77E 03	2.77E 03	2.77E 03	2.77E 03
CE	0.0	6.66E 03	6.54E 03	6.34E 03	6.21E 03	5.55E 03	5.47E 03
PR	0.0	2.59E 03	2.63E 03	2.70E 03	2.72E 03	2.73E 03	2.73E 03
ND	0.0	7.87E 03	8.01E 03	8.16E 03	8.28E 03	8.93E 03	9.00E 03
PM	0.0	8.90E 02	8.90E 02	8.54E 02	8.17E 02	4.11E 02	3.25E-01
SM	0.0	1.89E 03	1.92E 03	1.96E 03	1.99E 03	2.39E 03	2.73E 03
EU	0.0	3.67E 02	3.59E 02	3.48E 02	3.39E 02	2.57E 02	2.71E 02
GD	0.0	4.47E 02	4.56E 02	4.67E 02	4.76E 02	5.62E 02	6.24E 02
TB	0.0	1.01E 02	9.97E 01	9.87E 01	9.81E 01	9.74E 01	9.74E 01
DY	0.0	3.61E 01	3.76E 01	3.92E 01	4.02E 01	4.44E 01	4.51E 01
HO	0.0	5.76E-01	5.75E-01	5.75E-01	5.75E-01	5.75E-01	5.75E-01
ER	0.0	4.04E-01	4.08E-01	4.08E-01	4.08E-01	4.08E-01	4.08E-01
TOTALS	0.0	8.47E 04	8.47E 04	8.47E 04	8.47E 04	8.47E 04	8.47E 04

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H 3	0.0	2.34E 03	2.33E 03	2.31E 03	2.29E 03	1.98E 03	4.32E 02
KR 85	0.0	2.48E 04	2.47E 04	2.44E 04	2.42E 04	2.05E 04	3.59E 03
RB 86	0.0	8.15E 03	2.67E 03	2.88E 02	3.10E 01	1.76E-14	0.0
SR 89	0.0	2.24E 06	1.50E 06	6.74E 05	3.03E 05	1.01E 00	0.0
SR 90	0.0	1.03E 05	1.03E 05	1.02E 05	1.02E 05	9.56E 04	4.91E 04
Y 90	0.0	1.06E 05	1.03E 05	1.02E 05	1.02E 05	9.57E 04	4.91E 04
Y 91	0.0	3.08E 06	2.17E 06	1.07E 06	5.27E 05	7.60E 00	0.0
ZR 93	0.0	3.40E 00	3.40E 00	3.40E 00	3.40E 00	3.40E 00	3.40E 00
NB 93M	0.0	1.24E-01	1.38E-01	1.65E-01	1.92E-01	5.89E-01	2.69E 00
ZR 95	0.0	7.03E 06	5.11E 06	2.69E 06	1.42E 06	5.94E 01	0.0
NB 95M	0.0	1.41E 05	1.08E 05	5.74E 04	3.02E 04	1.26E 00	1.83E-12
NB 95	0.0	6.97E 06	6.52E 06	4.45E 06	2.66E 06	1.28E 02	0.0
MO 99	0.0	7.45E 06	4.34E 03	1.47E-03	4.99E-10	0.0	0.0
TC 99M	0.0	6.48E 06	4.15E 03	1.41E-03	4.77E-10	0.0	0.0
TC 99	0.0	3.55E 01	3.59E 01	3.59E 01	3.59E 01	3.59E 01	3.58E 01
RU103	0.0	7.16E 06	4.24E 06	1.48E 06	5.18E 05	3.34E-02	0.0
RH103M	0.0	7.16E 06	4.24E 06	1.48E 06	5.19E 05	3.34E-02	0.0
RH105	0.0	4.76E 06	4.37E 00	3.68E-12	0.0	0.0	0.0
RU106	0.0	3.42E 06	3.23E 06	2.89E 06	2.58E 06	4.32E 05	3.52E-03
RH106	0.0	3.42E 06	3.23E 06	2.89E 06	2.58E 06	4.32E 05	3.52E-03
AG110M	0.0	4.65E 03	4.28E 03	3.63E 03	3.08E 03	2.31E 02	4.25E-10
AG110	0.0	3.23E 05	5.57E 02	4.72E 02	4.01E 02	3.00E 01	5.52E-11
AG111	0.0	5.06E 05	3.17E 04	1.24E 02	4.83E-01	0.0	0.0
CD113M	0.0	3.41E 02	3.40E 02	3.37E 02	3.34E 02	2.94E 02	7.73E 01
IN114M	0.0	5.92E 00	3.90E 00	1.70E 00	7.39E-01	1.49E-06	0.0
IN114	0.0	1.01E 01	3.77E 00	1.64E 00	7.14E-01	1.44E-06	0.0
CD115M	0.0	9.52E 02	5.87E 02	2.23E 02	8.48E 01	2.02E-05	0.0
CD115	0.0	1.31E 04	1.17E 00	9.27E-09	7.32E-17	0.0	0.0
IN115M	0.0	1.31E 04	1.28E 00	1.01E-08	7.99E-17	0.0	0.0
SN117M	0.0	5.55E 00	1.26E 00	6.44E-02	3.30E-03	0.0	0.0
SN119M	0.0	5.01E 01	4.61E 01	3.90E 01	3.30E 01	2.40E 00	3.19E-12
SN121M	0.0	1.41E 02	1.41E 02	1.41E 02	1.41E 02	1.37E 02	1.07E 02
SB122	0.0	4.30E 03	2.56E 00	9.07E-07	3.21E-13	0.0	0.0
SN123M	0.0	2.06E 03	1.75E 03	1.25E 03	8.98E 02	4.73E 00	0.0
TE123M	0.0	9.49E 00	7.94E 00	5.57E 00	3.90E 00	1.44E-02	0.0
SB124	0.0	2.43E 02	1.72E 02	8.60E 01	4.30E 01	7.76E-04	0.0
SN125	0.0	1.53E 05	1.67E 04	2.00E 02	2.40E 00	0.0	0.0
SB125	0.0	5.07E 04	5.09E 04	4.89E 04	4.69E 04	2.41E 04	2.35E 01
TE125M	0.0	1.70E 04	1.79E 04	1.87E 04	1.87E 04	9.90E 03	9.74E 00
SN126	0.0	4.40E 00	4.40E 00	4.40E 00	4.40E 00	4.40E 00	4.40E 00
SB126M	0.0	5.05E 03	4.40E 00	4.40E 00	4.40E 00	4.40E 00	4.40E 00
SB126	0.0	1.02E 04	1.94E 03	7.38E 01	6.85E 00	4.36E 00	4.36E 00
SB127	0.0	8.46E 05	4.04E 03	8.88E-02	1.95E-06	0.0	0.0
TE127M	0.0	1.82E 05	1.56E 05	1.07E 05	7.29E 04	1.78E 02	0.0
TE127	0.0	8.39E 05	1.58E 05	1.05E 05	7.20E 04	1.76E 02	1.53E-05
TE129M	0.0	8.17E 05	4.46E 05	1.31E 05	3.86E 04	1.63E-04	0.0
TE129	0.0	1.96E 06	2.86E 05	8.41E 04	2.47E 04	1.04E-04	0.0

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
I129	0.0	1.30E-01	1.32E-01	1.34E-01	1.34E-01	1.34E-01	1.34E-01
I131	0.0	4.32E 06	3.36E 05	1.92E 03	1.09E 01	0.0	0.0
XE131M	0.0	4.52E 04	1.51E 04	5.81E 02	1.79E 01	0.0	0.0
TE132	0.0	6.06E 06	1.01E 04	2.88E-02	8.20E-08	0.0	0.0
I132	0.0	6.08E 06	1.04E 04	2.96E-02	8.45E-08	0.0	0.0
XE133M	0.0	1.98E 05	2.97E 01	3.03E-07	3.09E-15	0.0	0.0
XE133	0.0	7.91E 06	1.81E 05	6.75E 01	2.52E-02	0.0	0.0
CS134	0.0	7.86E 04	7.65E 04	7.24E 04	6.84E 04	2.85E 04	3.16E 00
CS135	0.0	2.87E 00	2.87E 00	2.87E 00	2.87E 00	2.87E 00	2.87E 00
CS136	0.0	3.65E 05	7.38E 04	3.01E 03	1.23E 02	0.0	0.0
CS137	0.0	2.66E 05	2.66E 05	2.65E 05	2.64E 05	2.48E 05	1.33E 05
BA137M	0.0	2.49E 05	2.48E 05	2.47E 05	2.45E 05	2.32E 05	1.24E 05
BA140	0.0	6.36E 06	1.25E 06	4.86E 04	1.89E 03	0.0	0.0
LA140	0.0	6.60E 06	1.44E 06	5.60E 04	2.17E 03	2.17E-15	0.0
CE141	0.0	6.75E 06	3.57E 06	9.89E 05	2.74E 05	4.45E-04	0.0
CE143	0.0	6.36E 06	1.73E 00	1.27E-13	0.0	0.0	0.0
PR143	0.0	6.38E 06	1.85E 06	8.90E 04	4.27E 03	6.99E-18	0.0
CE144	0.0	3.32E 06	3.09E 06	2.67E 06	2.31E 06	2.29E 05	8.07E-06
PR144	0.0	3.48E 06	3.09E 06	2.67E 06	2.31E 06	2.29E 05	8.07E-06
ND147	0.0	2.83E 06	4.34E 05	1.02E 04	2.42E 02	0.0	0.0
PM147	0.0	8.12E 05	8.22E 05	7.92E 05	7.58E 05	3.82E 05	3.02E 02
PM148M	0.0	1.77E 05	1.08E 05	4.00E 04	1.49E 04	2.47E-03	0.0
PM148	0.0	2.10E 05	1.28E 04	3.22E 03	1.19E 03	1.98E-04	0.0
PM149	0.0	1.70E 06	1.46E 02	1.00E-06	6.89E-15	0.0	0.0
SM151	0.0	1.09E 04	1.09E 04	1.09E 04	1.09E 04	1.07E 04	8.62E 03
EU152	0.0	2.58E 01	2.57E 01	2.54E 01	2.52E 01	2.17E 01	4.56E 00
SM153	0.0	4.70E 05	1.16E 01	6.94E-09	0.0	0.0	0.0
GD153	0.0	1.45E 00	1.33E 00	1.12E 00	9.44E-01	6.28E-02	3.39E-14
EU154	0.0	2.53E 03	2.52E 03	2.50E 03	2.49E 03	2.22E 03	6.90E 02
EU155	0.0	2.09E 05	2.02E 05	1.90E 05	1.78E 05	6.62E 04	2.14E 00
EU156	0.0	3.04E 05	7.76E 04	4.85E 03	3.03E 02	3.14E-17	0.0
TB160	0.0	3.41E 04	2.56E 04	1.44E 04	8.07E 03	9.06E-01	0.0
TB161	0.0	4.74E 04	2.33E 03	5.61E 00	1.35E-02	0.0	0.0
GD162	0.0	1.21E 04	1.15E 04	1.02E 04	9.13E 03	1.52E 03	1.13E-05
TB162M	0.0	1.21E 04	1.15E 04	1.02E 04	9.13E 03	1.52E 03	1.13E-05
DY166	0.0	7.10E 02	1.55E 00	7.56E-06	3.68E-11	0.0	0.0
HO166	0.0	9.56E 02	2.32E 00	1.13E-05	5.49E-11	0.0	0.0
SUBTOT	0.0	1.37E 08	4.90E 07	2.66E 07	1.81E 07	2.54E 06	3.70E 05
TOTALS	0.0	6.21E 08	4.90E 07	2.66E 07	1.81E 07	2.54E 06	3.70E 05

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
KR 85	0.0	3.99E 01	3.97E 01	3.92E 01	3.88E 01	3.29E 01	5.77E 00
RB 86	0.0	3.83E 01	1.26E 01	1.35E 00	1.46E-01	8.26E-17	0.0
SR 89	0.0	8.04E 03	5.39E 03	2.42E 03	1.09E 03	3.64E-03	0.0
SR 90	0.0	1.34E 02	1.34E 02	1.33E 02	1.33E 02	1.25E 02	6.41E 01
Y 90	0.0	6.09E 02	5.88E 02	5.86E 02	5.83E 02	5.47E 02	2.81E 02
Y 91	0.0	1.17E 04	8.25E 03	4.07E 03	2.01E 03	2.89E-02	0.0
ZR 95	0.0	3.68E 04	2.67E 04	1.41E 04	7.44E 03	3.11E-01	0.0
NB 95M	0.0	1.96E 02	1.51E 02	8.00E 01	4.20E 01	1.76E-03	2.55E-15
NB 95	0.0	3.34E 04	3.13E 04	2.14E 04	1.28E 04	6.16E-01	0.0
MO 99	0.0	3.35E 04	1.95E 01	6.61E-06	2.24E-12	0.0	0.0
TC 99M	0.0	5.50E 03	3.52E 00	1.19E-06	4.04E-13	0.0	0.0
RU103	0.0	2.34E 04	1.38E 04	4.84E 03	1.69E 03	1.09E-04	0.0
RH103M	0.0	1.70E 03	1.01E 03	3.52E 02	1.23E 02	7.92E-06	0.0
RU106	0.0	2.03E 02	1.92E 02	1.71E 02	1.53E 02	2.56E 01	2.08E-07
RH106	0.0	3.31E 04	3.12E 04	2.79E 04	2.49E 04	4.17E 03	3.40E-05
AG110M	0.0	7.50E 01	6.91E 01	5.86E 01	4.97E 01	3.72E 00	6.85E-12
AG110	0.0	2.34E 03	4.04E 00	3.43E 00	2.91E 00	2.18E-01	4.01E-13
AG111	0.0	1.21E 03	7.58E 01	2.96E-01	1.16E-03	0.0	0.0
CD113M	0.0	4.51E-01	4.49E-01	4.46E-01	4.42E-01	3.89E-01	1.02E-01
CD115M	0.0	3.87E 00	2.39E 00	9.08E-01	3.45E-01	8.23E-08	0.0
SN121M	0.0	1.48E-01	1.48E-01	1.48E-01	1.48E-01	1.44E-01	1.13E-01
SN123M	0.0	7.03E 00	5.96E 00	4.27E 00	3.06E 00	1.61E-02	0.0
SB124	0.0	3.29E 00	2.33E 00	1.16E 00	5.81E-01	1.05E-05	0.0
SN125	0.0	9.25E 02	1.01E 02	1.21E 00	1.45E-02	0.0	0.0
SB125	0.0	1.70E 02	1.71E 02	1.64E 02	1.57E 02	8.09E 01	7.89E-02
TE125M	0.0	1.46E 01	1.54E 01	1.61E 01	1.60E 01	8.51E 00	8.37E-03
SB126	0.0	1.12E 02	2.13E 01	8.09E-01	7.51E-02	4.78E-02	4.78E-02
SB127	0.0	4.65E 03	2.22E 01	4.88E-04	1.07E-08	0.0	0.0
TE127M	0.0	1.00E 02	8.61E 01	5.88E 01	4.02E 01	9.80E-02	0.0
TE127	0.0	1.37E 03	2.57E 02	1.72E 02	1.17E 02	2.87E-01	2.50E-08
TE129M	0.0	1.62E 03	8.82E 02	2.60E 02	7.64E 01	3.22E-07	0.0
TE129	0.0	7.13E 03	1.04E 03	3.05E 02	8.97E 01	3.78E-07	0.0
I131	0.0	1.78E 04	1.38E 03	7.89E 00	4.50E-02	0.0	0.0
XE131M	0.0	4.39E 01	1.47E 01	5.65E-01	1.74E-02	0.0	0.0
TE132	0.0	8.26E 03	1.38E 01	3.92E-05	1.12E-10	0.0	0.0
I132	0.0	9.73E 04	1.67E 02	4.75E-04	1.35E-09	0.0	0.0
XE133	0.0	8.54E 03	1.95E 02	7.28E-02	2.72E-05	0.0	0.0
CS134	0.0	8.25E 02	8.02E 02	7.59E 02	7.18E 02	2.99E 02	3.31E-02
CS136	0.0	5.65E 03	1.14E 03	4.66E 01	1.90E 00	0.0	0.0
CS137	0.0	4.31E 02	4.30E 02	4.28E 02	4.26E 02	4.02E 02	2.15E 02
BA137M	0.0	9.76E 02	9.74E 02	9.71E 02	9.67E 02	9.11E 02	4.88E 02
BA140	0.0	2.15E 04	4.23E 03	1.64E 02	6.36E 00	0.0	0.0
LA140	0.0	1.10E 05	2.39E 04	9.29E 02	3.60E 01	3.60E-17	0.0
CE141	0.0	1.33E 04	7.02E 03	1.95E 03	5.39E 02	8.76E-07	0.0
PR143	0.0	1.38E 04	4.02E 03	1.93E 02	9.27E 00	1.52E-20	0.0
CE144	0.0	2.92E 03	2.71E 03	2.34E 03	2.02E 03	2.01E 02	7.08E-09
PR144	0.0	2.59E 04	2.30E 04	1.99E 04	1.72E 04	1.70E 03	6.00E-08

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
ND147	0.0	7.91E 03	1.22E 03	2.87E 01	6.77E-01	0.0	0.0
PM147	0.0	4.19E 02	4.24E 02	4.08E 02	3.91E 02	1.97E 02	1.56E-01
PM148M	0.0	2.24E 03	1.37E 03	5.08E 02	1.89E 02	3.13E-05	0.0
PM148	0.0	1.72E 03	1.05E 02	2.63E 01	9.76E 00	1.62E-06	0.0
PM149	0.0	4.25E 03	3.65E-01	2.50E-09	1.72E-17	0.0	0.0
SM151	0.0	1.90E 01	1.91E 01	1.91E 01	1.90E 01	1.86E 01	1.50E 01
EU152	0.0	3.15E-01	3.13E-01	3.10E-01	3.07E-01	2.65E-01	5.56E-02
EU154	0.0	2.37E 01	2.36E 01	2.35E 01	2.33E 01	2.08E 01	6.46E 00
EU155	0.0	1.76E 02	1.70E 02	1.60E 02	1.50E 02	5.57E 01	1.80E-03
EU156	0.0	3.21E 03	8.19E 02	5.12E 01	3.20E 00	3.31E-19	0.0
TB160	0.0	2.87E 02	2.15E 02	1.21E 02	6.79E 01	7.62E-03	0.0
TB161	0.0	7.73E 01	3.80E 00	9.15E-03	2.21E-05	0.0	0.0
GD162	0.0	4.13E 01	3.90E 01	3.48E 01	3.11E 01	5.16E 00	3.84E-08
TB162M	0.0	8.13E 01	7.68E 01	6.86E 01	6.12E 01	1.02E 01	7.56E-08
SUBTOT	0.0	5.55E 05	1.96E 05	1.06E 05	7.43E 04	8.82E 03	1.08E 03
TOTALS	0.0	6.35E 06	1.96E 05	1.06E 05	7.43E 04	8.82E 03	1.08E 03

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
KR 85	0.0	3.19E-01	3.17E-01	3.14E-01	3.11E-01	2.63E-01	4.62E-02
RB 86	0.0	4.45E 00	1.46E 00	1.57E-01	1.69E-02	9.58E-18	0.0
Y 91	0.0	7.03E 01	4.95E 01	2.44E 01	1.20E 01	1.74E-04	0.0
ZR 95	0.0	3.02E 04	2.20E 04	1.16E 04	6.11E 03	2.55E-01	0.0
NB 95M	0.0	1.96E 02	1.51E 02	8.00E 01	4.20E 01	1.76E-03	2.55E-15
NB 95	0.0	3.16E 04	2.95E 04	2.02E 04	1.21E 04	5.82E-01	0.0
MO 99	0.0	9.71E 03	5.66E 00	1.92E-06	6.50E-13	0.0	0.0
RU103	0.0	2.06E 04	1.22E 04	4.26E 03	1.49E 03	9.59E-05	0.0
RH106	0.0	3.97E 03	3.75E 03	3.35E 03	2.99E 03	5.01E 02	4.08E-06
AG110M	0.0	7.20E 01	6.63E 01	5.62E 01	4.77E 01	3.57E 00	6.58E-12
AG110	0.0	6.33E 01	1.09E-01	9.25E-02	7.85E-02	5.88E-03	1.08E-14
CD115M	0.0	5.70E-01	3.51E-01	1.34E-01	5.08E-02	1.21E-08	0.0
SB124	0.0	2.70E 00	1.91E 00	9.53E-01	4.77E-01	8.60E-06	0.0
SN125	0.0	8.42E 01	9.21E 00	1.10E-01	1.32E-03	0.0	0.0
SB125	0.0	9.35E 01	9.39E 01	9.02E 01	8.65E 01	4.45E 01	4.34E-02
SB126	0.0	6.49E 01	1.23E 01	4.69E-01	4.36E-02	2.77E-02	2.77E-02
SB127	0.0	2.36E 03	1.13E 01	2.47E-04	5.43E-09	0.0	0.0
TE127	0.0	2.05E 01	3.86E 00	2.58E 00	1.76E 00	4.30E-03	3.75E-10
TE129M	0.0	1.94E 02	1.06E 02	3.12E 01	9.17E 00	3.86E-08	0.0
TE129	0.0	1.07E 03	1.55E 02	4.57E 01	1.35E 01	5.67E-08	0.0
I131	0.0	1.02E 04	7.94E 02	4.53E 00	2.58E-02	0.0	0.0
TE132	0.0	3.14E 03	5.23E 00	1.49E-05	4.25E-11	0.0	0.0
I132	0.0	7.59E 04	1.30E 02	3.70E-04	1.05E-09	0.0	0.0
CS134	0.0	7.34E 02	7.14E 02	6.76E 02	6.39E 02	2.66E 02	2.95E-02
CS136	0.0	5.31E 03	1.07E 03	4.38E 01	1.78E 00	0.0	0.0
BA137M	0.0	9.76E 02	9.74E 02	9.71E 02	9.67E 02	9.11E 02	4.88E 02
BA140	0.0	1.01E 04	1.99E 03	7.71E 01	2.99E 00	0.0	0.0
LA140	0.0	8.43E 04	1.84E 04	7.15E 02	2.78E 01	2.77E-17	0.0
CE144	0.0	7.87E 02	7.32E 02	6.32E 02	5.46E 02	5.43E 01	1.91E-09
PR144	0.0	6.21E 02	5.52E 02	4.77E 02	4.12E 02	4.09E 01	1.44E-09
ND147	0.0	3.40E 02	5.23E 01	1.23E 00	2.91E-02	0.0	0.0
PM148M	0.0	2.04E 03	1.24E 03	4.62E 02	1.72E 02	2.85E-05	0.0
PM148	0.0	7.92E 02	4.83E 01	1.21E 01	4.49E 00	7.45E-07	0.0
EU154	0.0	1.99E 01	1.98E 01	1.97E 01	1.96E 01	1.75E 01	5.43E 00
EU156	0.0	2.44E 03	6.23E 02	3.89E 01	2.43E 00	2.52E-19	0.0
TB160	0.0	2.36E 02	1.77E 02	9.92E 01	5.57E 01	6.25E-03	0.0
SUBTOT	0.0	2.98E 05	9.56E 04	4.39E 04	2.57E 04	1.84E 03	4.94E 02
TOTALS	0.0	2.21E 06	9.56E 04	4.39E 04	2.57E 04	1.84E 03	4.94E 02

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	8.33E-02	8.29E-02	8.22E-02	8.14E-02	7.03E-02	1.54E-02
ZN	0.0	1.91E 00	4.16E-05	1.98E-14	0.0	0.0	0.0
GA	0.0	3.54E 01	8.40E-04	4.00E-13	0.0	0.0	0.0
GE	0.0	1.12E 01	1.62E-19	0.0	0.0	0.0	0.0
AS	0.0	1.98E 02	2.01E-06	1.27E-17	0.0	0.0	0.0
SE	0.0	1.36E 04	2.17E-05	2.17E-05	2.17E-05	2.17E-05	2.17E-05
BR	0.0	1.28E 05	7.38E-06	4.00E-18	0.0	0.0	0.0
KR	0.0	1.93E 05	3.97E 01	3.92E 01	3.88E 01	3.29E 01	5.77E 00
RB	0.0	5.26E 05	1.26E 01	1.35E 00	1.46E-01	1.65E-08	1.65E-08
SR	0.0	3.27E 05	5.53E 03	2.56E 03	1.22E 03	1.25E 02	6.41E 01
Y	0.0	4.73E 05	8.84E 03	4.65E 03	2.59E 03	5.47E 02	2.81E 02
ZR	0.0	8.36E 04	2.67E 04	1.41E 04	7.44E 03	3.11E-01	4.03E-04
NB	0.0	4.22E 05	3.14E 04	2.14E 04	1.28E 04	6.18E-01	4.79E-04
MO	0.0	2.68E 05	1.95E 01	6.61E-06	2.24E-12	0.0	0.0
TC	0.0	4.05E 05	3.54E 00	2.42E-02	2.42E-02	2.42E-02	2.42E-02
RU	0.0	9.94E 04	1.40E 04	5.01E 03	1.85E 03	2.56E 01	2.08E-07
RH	0.0	1.23E 05	3.23E 04	2.83E 04	2.50E 04	4.17E 03	3.40E-05
PD	0.0	1.20E 04	7.19E-09	0.0	0.0	0.0	0.0
AG	0.0	1.01E 04	1.49E 02	6.23E 01	5.26E 01	3.94E 00	7.25E-12
CD	0.0	6.47E 02	2.84E 00	1.35E 00	7.87E-01	3.89E-01	1.02E-01
IN	0.0	1.26E 03	2.58E-02	1.01E-02	4.41E-03	8.89E-09	0.0
SN	0.0	2.78E 04	1.07E 02	5.65E 00	3.24E 00	1.64E-01	1.15E-01
SB	0.0	3.97E 05	2.17E 02	1.66E 02	1.58E 02	8.09E 01	1.29E-01
TE	0.0	3.01E 05	2.29E 03	8.11E 02	3.40E 02	8.89E 00	8.37E-03
I	0.0	6.18E 05	1.55E 03	7.89E 00	4.50E-02	5.81E-05	5.81E-05
XE	0.0	3.79E 05	2.10E 02	6.38E-01	1.74E-02	0.0	0.0
CS	0.0	6.20E 05	2.37E 03	1.23E 03	1.15E 03	7.01E 02	2.15E 02
BA	0.0	2.13E 05	5.20E 03	1.13E 03	9.73E 02	9.11E 02	4.88E 02
LA	0.0	3.82E 05	2.39E 04	9.29E 02	3.60E 01	3.60E-17	0.0
CE	0.0	1.16E 05	9.74E 03	4.29E 03	2.56E 03	2.01E 02	7.08E-09
PR	0.0	1.48E 05	2.70E 04	2.00E 04	1.72E 04	1.70E 03	6.00E-08
ND	0.0	2.40E 04	1.22E 03	2.87E 01	6.77E-01	0.0	0.0
PM	0.0	2.05E 04	1.90E 03	9.43E 02	5.89E 02	1.97E 02	1.56E-01
SM	0.0	3.58E 03	1.91E 01	1.91E 01	1.90E 01	1.86E 01	1.50E 01
EU	0.0	1.03E 04	1.01E 03	2.35E 02	1.77E 02	7.68E 01	6.52E 00
GD	0.0	8.92E 02	3.90E 01	3.48E 01	3.11E 01	5.16E 00	3.84E-08
TB	0.0	4.68E 02	2.96E 02	1.90E 02	1.29E 02	1.02E 01	7.56E-08
DY	0.0	5.22E 00	1.32E-03	6.41E-09	3.12E-14	0.0	0.0
HO	0.0	4.19E 00	1.02E-02	4.94E-08	2.40E-13	0.0	0.0
TOTALS	0.0	6.35E 06	1.96E 05	1.06E 05	7.43E 04	8.82E 03	1.08E 03

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
SE 82	0.0	1.58E 00	1.58E 00	1.58E 00	1.58E 00	1.58E 00	1.58E 00
KR 83	0.0	4.36E 00	4.36E 00	4.36E 00	4.36E 00	4.36E 00	4.36E 00
KR 84	0.0	8.14E 00	8.14E 00	8.14E 00	8.14E 00	8.14E 00	8.14E 00
KR 85	0.0	2.13E 00	2.12E 00	2.10E 00	2.08E 00	1.76E 00	3.09E-01
RB 85	0.0	4.44E 00	4.45E 00	4.48E 00	4.50E 00	4.82E 00	6.27E 00
KR 86	0.0	1.20E 01	1.20E 01	1.20E 01	1.20E 01	1.20E 01	1.20E 01
RB 87	0.0	1.56E 01	1.56E 01	1.56E 01	1.56E 01	1.56E 01	1.56E 01
SR 88	0.0	2.23E 01	2.23E 01	2.23E 01	2.23E 01	2.23E 01	2.23E 01
SR 89	0.0	4.87E 00	3.26E 00	1.47E 00	6.59E-01	2.20E-06	0.0
Y 89	0.0	2.14E 01	2.30E 01	2.48E 01	2.56E 01	2.63E 01	2.63E 01
SR 90	0.0	2.96E 01	2.96E 01	2.94E 01	2.93E 01	2.75E 01	1.41E 01
Y 91	0.0	7.66E 00	5.40E 00	2.66E 00	1.31E 00	1.89E-05	0.0
ZR 91	0.0	2.96E 01	3.19E 01	3.47E 01	3.60E 01	3.73E 01	3.73E 01
ZR 92	0.0	4.31E 01	4.32E 01	4.32E 01	4.32E 01	4.32E 01	4.32E 01
ZR 93	0.0	4.94E 01	4.95E 01	4.95E 01	4.95E 01	4.95E 01	4.95E 01
ZR 94	0.0	5.37E 01	5.37E 01	5.37E 01	5.37E 01	5.37E 01	5.37E 01
ZR 95	0.0	1.42E 01	1.03E 01	5.43E 00	2.86E 00	1.20E-04	0.0
NB 95	0.0	7.07E 00	6.79E 00	4.75E 00	2.86E 00	1.39E-04	0.0
MO 95	0.0	3.89E 01	4.31E 01	5.00E 01	5.44E 01	6.01E 01	6.01E 01
ZR 96	0.0	6.13E 01	6.13E 01	6.13E 01	6.13E 01	6.13E 01	6.13E 01
MO 97	0.0	6.76E 01	6.78E 01	6.78E 01	6.78E 01	6.78E 01	6.78E 01
MO 98	0.0	6.70E 01	6.70E 01	6.70E 01	6.70E 01	6.70E 01	6.70E 01
TC 99	0.0	6.87E 01	6.95E 01	6.95E 01	6.95E 01	6.95E 01	6.95E 01
MO100	0.0	7.28E 01	7.28E 01	7.28E 01	7.28E 01	7.28E 01	7.28E 01
RU101	0.0	7.09E 01	7.09E 01	7.09E 01	7.09E 01	7.09E 01	7.09E 01
RU102	0.0	7.40E 01	7.40E 01	7.40E 01	7.40E 01	7.40E 01	7.40E 01
RU103	0.0	1.09E 01	6.47E 00	2.26E 00	7.92E-01	5.10E-08	0.0
RH103	0.0	6.25E 01	6.69E 01	7.11E 01	7.26E 01	7.34E 01	7.34E 01
RU104	0.0	6.01E 01	6.01E 01	6.01E 01	6.01E 01	6.01E 01	6.01E 01
PD105	0.0	3.97E 01	4.00E 01	4.00E 01	4.00E 01	4.00E 01	4.00E 01
RU106	0.0	2.53E 01	2.39E 01	2.14E 01	1.91E 01	3.20E 00	2.60E-08
PD106	0.0	1.14E 01	1.28E 01	1.54E 01	1.77E 01	3.36E 01	3.67E 01
PD107	0.0	2.76E 01	2.76E 01	2.76E 01	2.76E 01	2.76E 01	2.76E 01
PD108	0.0	1.12E 01	1.12E 01	1.12E 01	1.12E 01	1.12E 01	1.12E 01
AG109	0.0	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01
PD110	0.0	1.44E 00	1.44E 00	1.44E 00	1.44E 00	1.44E 00	1.44E 00
CD111	0.0	1.92E 00	1.99E 00	2.00E 00	2.00E 00	2.00E 00	2.00E 00
CD112	0.0	1.10E 00	1.11E 00	1.11E 00	1.11E 00	1.11E 00	1.11E 00
SN126	0.0	1.59E 00	1.59E 00	1.59E 00	1.59E 00	1.59E 00	1.59E 00
I127	0.0	3.30E 00	3.43E 00	3.52E 00	3.58E 00	3.72E 00	3.72E 00
TE128	0.0	7.24E 00	7.24E 00	7.24E 00	7.24E 00	7.24E 00	7.24E 00
I129	0.0	1.95E 01	1.99E 01	2.03E 01	2.05E 01	2.05E 01	2.05E 01
TE130	0.0	2.87E 01	2.87E 01	2.87E 01	2.87E 01	2.87E 01	2.87E 01
XE131	0.0	4.78E 01	4.93E 01	4.94E 01	4.94E 01	4.94E 01	4.94E 01
XE132	0.0	7.03E 01	7.12E 01	7.12E 01	7.12E 01	7.12E 01	7.12E 01
CS133	0.0	8.45E 01	8.68E 01	8.69E 01	8.69E 01	8.69E 01	8.69E 01
XE134	0.0	1.00E 02	1.00E 02	1.00E 02	1.00E 02	1.00E 02	1.00E 02

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
CS135	0.0	9.32E 01	9.32E 01	9.32E 01	9.32E 01	9.32E 01	9.32E 01
XE136	0.0	9.05E 01	9.05E 01	9.05E 01	9.05E 01	9.05E 01	9.05E 01
BA136	0.0	1.65E 00	1.73E 00	1.75E 00	1.75E 00	1.75E 00	1.75E 00
CS137	0.0	9.40E 01	9.38E 01	9.35E 01	9.31E 01	8.77E 01	4.70E 01
BA137	0.0	1.31E 00	1.49E 00	1.85E 00	2.20E 00	7.61E 00	4.83E 01
BA138	0.0	9.24E 01	9.24E 01	9.24E 01	9.24E 01	9.24E 01	9.24E 01
LA139	0.0	8.95E 01	8.96E 01	8.96E 01	8.96E 01	8.96E 01	8.96E 01
CE140	0.0	9.67E 01	1.01E 02	1.02E 02	1.02E 02	1.02E 02	1.02E 02
CE141	0.0	1.14E 01	6.01E 00	1.66E 00	4.61E-01	7.49E-10	0.0
PR141	0.0	7.98E 01	8.52E 01	8.95E 01	9.07E 01	9.12E 01	9.12E 01
CE142	0.0	8.67E 01	8.68E 01	8.68E 01	8.68E 01	8.68E 01	8.68E 01
PR143	0.0	4.45E 00	1.09E 00	5.22E-02	2.51E-03	0.0	0.0
ND143	0.0	7.64E 01	8.03E 01	8.13E 01	8.14E 01	8.14E 01	8.14E 01
CE144	0.0	4.27E 01	3.97E 01	3.43E 01	2.96E 01	2.94E 00	1.04E-10
ND144	0.0	2.74E 01	3.05E 01	3.59E 01	4.05E 01	6.72E 01	7.02E 01
ND145	0.0	7.07E 01	7.07E 01	7.07E 01	7.07E 01	7.07E 01	7.07E 01
ND146	0.0	6.30E 01	6.30E 01	6.30E 01	6.30E 01	6.30E 01	6.30E 01
PM147	0.0	4.25E 01	4.34E 01	4.19E 01	4.01E 01	2.02E 01	1.60E-02
SM147	0.0	7.06E 00	8.00E 00	9.87E 00	1.16E 01	3.15E 01	5.18E 01
ND148	0.0	3.81E 01	3.81E 01	3.81E 01	3.81E 01	3.81E 01	3.81E 01
SM149	0.0	2.77E 01	2.80E 01	2.80E 01	2.80E 01	2.80E 01	2.80E 01
ND150	0.0	2.31E 01	2.31E 01	2.31E 01	2.31E 01	2.31E 01	2.31E 01
SM151	0.0	1.83E 01	1.84E 01	1.83E 01	1.83E 01	1.80E 01	1.45E 01
SM152	0.0	1.37E 01	1.37E 01	1.37E 01	1.37E 01	1.37E 01	1.37E 01
EU153	0.0	6.63E 00	6.69E 00	6.69E 00	6.69E 00	6.69E 00	6.69E 00
SM154	0.0	4.37E 00	4.37E 00	4.37E 00	4.37E 00	4.37E 00	4.37E 00
EU155	0.0	2.76E 00	2.67E 00	2.51E 00	2.36E 00	8.74E-01	2.82E-05
GD156	0.0	1.66E 00	1.76E 00	1.79E 00	1.79E 00	1.79E 00	1.79E 00
GD157	0.0	1.01E 00	1.01E 00	1.01E 00	1.01E 00	1.01E 00	1.01E 00
GD158	0.0	1.62E 00	1.62E 00	1.62E 00	1.62E 00	1.62E 00	1.62E 00
SUBTOT	0.0	2.69E 03	2.70E 03	2.70E 03	2.70E 03	2.70E 03	2.68E 03
TOTALS	0.0	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	5.27E-03	5.25E-03	5.20E-03	5.15E-03	4.45E-03	9.72E-04
ZN	0.0	3.03E-05	6.62E-10	3.15E-19	0.0	0.0	0.0
GA	0.0	1.02E-05	2.88E-10	1.37E-19	0.0	0.0	0.0
GE	0.0	2.26E-02	2.26E-02	2.26E-02	2.26E-02	2.26E-02	2.26E-02
AS	0.0	7.82E-03	7.65E-03	7.65E-03	7.65E-03	7.65E-03	7.65E-03
SE	0.0	2.42E 00	2.42E 00	2.42E 00	2.42E 00	2.42E 00	2.42E 00
BR	0.0	8.33E-01	8.31E-01	8.31E-01	8.31E-01	8.31E-01	8.31E-01
KR	0.0	2.66E 01	2.66E 01	2.66E 01	2.65E 01	2.62E 01	2.48E 01
RB	0.0	2.01E 01	2.01E 01	2.01E 01	2.01E 01	2.04E 01	2.19E 01
SR	0.0	5.68E 01	5.51E 01	5.32E 01	5.23E 01	4.98E 01	3.64E 01
Y	0.0	2.92E 01	2.84E 01	2.75E 01	2.69E 01	2.63E 01	2.63E 01
ZR	0.0	2.52E 02	2.51E 02	2.49E 02	2.47E 02	2.48E 02	2.61E 02
NB	0.0	7.10E 00	6.80E 00	4.75E 00	2.86E 00	2.22E-04	7.00E-04
MO	0.0	2.47E 02	2.51E 02	2.58E 02	2.62E 02	2.68E 02	2.68E 02
TC	0.0	6.87E 01	6.95E 01	6.95E 01	6.95E 01	6.95E 01	6.95E 01
RU	0.0	2.42E 02	2.36E 02	2.29E 02	2.25E 02	2.08E 02	2.05E 02
RH	0.0	6.27E 01	6.69E 01	7.11E 01	7.26E 01	7.34E 01	7.34E 01
PD	0.0	9.15E 01	9.31E 01	9.57E 01	9.80E 01	1.14E 02	1.17E 02
AG	0.0	1.03E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01
CD	0.0	4.34E 00	4.42E 00	4.42E 00	4.42E 00	4.42E 00	4.42E 00
IN	0.0	3.72E-01	3.76E-01	3.77E-01	3.78E-01	3.79E-01	3.80E-01
SN	0.0	4.54E 00	4.50E 00	4.49E 00	4.49E 00	4.49E 00	4.48E 00
SB	0.0	2.25E 00	2.15E 00	2.11E 00	2.07E 00	1.63E 00	1.17E 00
TE	0.0	3.89E 01	3.74E 01	3.70E 01	3.68E 01	3.71E 01	3.76E 01
I	0.0	2.48E 01	2.35E 01	2.39E 01	2.40E 01	2.42E 01	2.42E 01
XE	0.0	3.11E 02	3.11E 02	3.12E 02	3.12E 02	3.12E 02	3.12E 02
CS	0.0	2.72E 02	2.74E 02	2.74E 02	2.73E 02	2.68E 02	2.27E 02
BA	0.0	9.98E 01	9.66E 01	9.61E 01	9.65E 01	1.02E 02	1.43E 02
LA	0.0	9.02E 01	8.97E 01	8.96E 01	8.96E 01	8.96E 01	8.96E 01
CE	0.0	2.38E 02	2.33E 02	2.24E 02	2.18E 02	1.91E 02	1.88E 02
PR	0.0	8.43E 01	8.63E 01	8.96E 01	9.07E 01	9.12E 01	9.12E 01
ND	0.0	3.01E 02	3.06E 02	3.12E 02	3.17E 02	3.44E 02	3.47E 02
PM	0.0	4.29E 01	4.35E 01	4.19E 01	4.01E 01	2.02E 01	1.60E-02
SM	0.0	7.23E 01	7.36E 01	7.55E 01	7.72E 01	9.68E 01	1.13E 02
EU	0.0	9.75E 00	9.63E 00	9.46E 00	9.33E 00	8.21E 00	1.07E 01
GD	0.0	5.06E 00	5.24E 00	5.43E 00	5.58E 00	7.04E 00	7.99E 00
TB	0.0	8.79E-01	8.75E-01	8.73E-01	8.72E-01	8.71E-01	8.71E-01
DY	0.0	2.41E-01	2.52E-01	2.60E-01	2.66E-01	3.04E-01	3.11E-01
HO	0.0	5.20E-03	5.18E-03	5.18E-03	5.18E-03	5.18E-03	5.18E-03
ER	0.0	3.02E-03	3.10E-03	3.10E-03	3.10E-03	3.10E-03	3.10E-03
TOTALS	0.0	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H 3	0.0	5.11E 01	5.09E 01	5.04E 01	4.99E 01	4.32E 01	9.43E 00
SE 79	0.0	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.48E-02	1.48E-02
KR 85	0.0	8.34E 02	8.30E 02	8.21E 02	8.12E 02	6.88E 02	1.21E 02
RB 86	0.0	7.69E 01	2.52E 01	2.72E 00	2.93E-01	1.66E-16	0.0
SR 89	0.0	1.37E 05	9.21E 04	4.14E 04	1.86E 04	6.21E-02	0.0
SR 90	0.0	4.19E 03	4.18E 03	4.16E 03	4.15E 03	3.89E 03	2.00E 03
Y 90	0.0	4.22E 03	4.18E 03	4.16E 03	4.15E 03	3.89E 03	2.00E 03
Y 91	0.0	1.87E 05	1.32E 05	6.50E 04	3.20E 04	4.62E-01	0.0
ZR 93	0.0	1.27E-01	1.27E-01	1.27E-01	1.27E-01	1.27E-01	1.27E-01
NB 93M	0.0	3.85E-03	4.37E-03	5.39E-03	6.40E-03	2.13E-02	1.00E-01
ZR 95	0.0	3.00E 05	2.18E 05	1.15E 05	6.06E 04	2.53E 00	0.0
NB 95M	0.0	6.00E 03	4.62E 03	2.44E 03	1.29E 03	5.38E-02	7.82E-14
NB 95	0.0	2.78E 05	2.67E 05	1.86E 05	1.12E 05	5.47E 00	0.0
MO 99	0.0	3.67E 05	2.14E 02	7.24E-05	2.45E-11	0.0	0.0
TC 99M	0.0	3.19E 05	2.04E 02	6.92E-05	2.35E-11	0.0	0.0
TC 99	0.0	1.17E 00	1.19E 00	1.19E 00	1.19E 00	1.19E 00	1.19E 00
RU103	0.0	3.50E 05	2.07E 05	7.25E 04	2.54E 04	1.63E-03	0.0
RH103M	0.0	3.50E 05	2.07E 05	7.26E 04	2.54E 04	1.63E-03	0.0
RH105	0.0	2.07E 05	1.90E-01	1.60E-13	0.0	0.0	0.0
RU106	0.0	8.51E 04	8.04E 04	7.18E 04	6.41E 04	1.07E 04	8.74E-05
RH106	0.0	8.51E 04	8.04E 04	7.18E 04	6.41E 04	1.07E 04	8.74E-05
AG110M	0.0	8.64E 00	7.96E 00	6.75E 00	5.73E 00	4.29E-01	7.89E-13
AG110	0.0	7.81E 02	1.03E 00	8.78E-01	7.45E-01	5.57E-02	1.03E-13
AG111	0.0	1.23E 04	7.71E 02	3.01E 00	1.18E-02	0.0	0.0
CD113M	0.0	6.14E-01	6.12E-01	6.07E-01	6.02E-01	5.29E-01	1.39E-01
IN114M	0.0	1.72E-03	1.13E-03	4.93E-04	2.15E-04	4.32E-10	0.0
IN114	0.0	3.02E-03	1.09E-03	4.76E-04	2.07E-04	4.17E-10	0.0
CD115M	0.0	1.01E 02	6.21E 01	2.36E 01	8.97E 00	2.14E-06	0.0
CD115	0.0	1.49E 03	1.33E-01	1.05E-09	8.31E-18	0.0	0.0
IN115M	0.0	1.49E 03	1.45E-01	1.15E-09	9.08E-18	0.0	0.0
SN117M	0.0	2.15E-02	4.86E-03	2.49E-04	1.28E-05	0.0	0.0
SN119M	0.0	3.50E 00	3.22E 00	2.72E 00	2.31E 00	1.67E-01	2.23E-13
SN121M	0.0	1.06E 00	1.06E 00	1.06E 00	1.06E 00	1.03E 00	8.08E-01
SB122	0.0	2.06E 01	1.23E-02	4.35E-09	1.54E-15	0.0	0.0
SN123M	0.0	4.10E 01	3.47E 01	2.49E 01	1.78E 01	9.40E-02	0.0
TE123M	0.0	6.48E-03	5.42E-03	3.80E-03	2.66E-03	9.80E-06	0.0
SB124	0.0	1.07E 01	7.54E 00	3.77E 00	1.88E 00	3.40E-05	0.0
SN125	0.0	4.33E 03	4.74E 02	5.68E 00	6.80E-02	0.0	0.0
SB125	0.0	1.03E 03	1.04E 03	1.00E 03	9.63E 02	4.95E 02	4.83E-01
TE125M	0.0	3.12E 02	3.41E 02	3.70E 02	3.77E 02	2.05E 02	2.00E-01
SN126	0.0	4.51E-02	4.51E-02	4.51E-02	4.51E-02	4.51E-02	4.51E-02
SB126M	0.0	1.53E 03	4.51E-02	4.51E-02	4.51E-02	4.51E-02	4.51E-02
SB126	0.0	1.54E 03	2.92E 02	1.05E 01	4.21E-01	4.46E-02	4.46E-02
SB127	0.0	2.03E 04	9.72E 01	2.13E-03	4.69E-08	0.0	0.0
TE127M	0.0	3.16E 03	2.75E 03	1.88E 03	1.28E 03	3.13E 00	0.0
TE127	0.0	1.90E 04	2.80E 03	1.86E 03	1.27E 03	3.10E 00	2.70E-07
TE129M	0.0	3.04E 04	1.66E 04	4.88E 03	1.44E 03	6.05E-06	0.0

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TE129	0.0	7.66E 04	1.06E 04	3.13E 03	9.20E 02	3.87E-06	0.0
I129	0.0	3.18E-03	3.25E-03	3.32E-03	3.34E-03	3.34E-03	3.34E-03
TE131M	0.0	3.01E 04	1.82E-03	0.0	0.0	0.0	0.0
I131	0.0	1.97E 05	1.53E 04	8.73E 01	4.98E-01	0.0	0.0
XE131M	0.0	1.74E 03	6.32E 02	2.49E 01	7.63E-01	0.0	0.0
TE132	0.0	2.79E 05	4.64E 02	1.32E-03	3.77E-09	0.0	0.0
I132	0.0	2.84E 05	4.79E 02	1.36E-03	3.89E-09	0.0	0.0
XE133M	0.0	7.86E 03	1.29E 00	1.32E-08	1.34E-16	0.0	0.0
XE133	0.0	3.47E 05	8.10E 03	3.03E 00	1.13E-03	0.0	0.0
CS134	0.0	3.97E 02	3.86E 02	3.66E 02	3.46E 02	1.44E 02	1.60E-02
CS135	0.0	8.23E-02	8.23E-02	8.23E-02	8.23E-02	8.23E-02	8.23E-02
CS136	0.0	7.39E 03	1.49E 03	6.09E 01	2.48E 00	0.0	0.0
CS137	0.0	8.18E 03	8.16E 03	8.13E 03	8.10E 03	7.63E 03	4.09E 03
BA137M	0.0	7.65E 03	7.63E 03	7.60E 03	7.58E 03	7.14E 03	3.82E 03
BA140	0.0	3.20E 05	6.31E 04	2.45E 03	9.50E 01	0.0	0.0
LA140	0.0	3.24E 05	7.26E 04	2.82E 03	1.09E 02	1.09E-16	0.0
CE141	0.0	3.25E 05	1.72E 05	4.76E 04	1.32E 04	2.14E-05	0.0
CE143	0.0	3.02E 05	8.22E-02	6.01E-15	0.0	0.0	0.0
PR143	0.0	2.97E 05	7.25E 04	3.48E 03	1.67E 02	0.0	0.0
CE144	0.0	1.36E 05	1.27E 05	1.09E 05	9.45E 04	9.39E 03	3.31E-07
PR144	0.0	1.39E 05	1.27E 05	1.09E 05	9.45E 04	9.39E 03	3.31E-07
ND147	0.0	1.77E 05	2.72E 04	6.42E 02	1.51E 01	0.0	0.0
PM147	0.0	3.95E 04	4.03E 04	3.89E 04	3.73E 04	1.88E 04	1.48E 01
PM148M	0.0	1.97E 03	1.20E 03	4.45E 02	1.65E 02	2.74E-05	0.0
PM148	0.0	2.31E 03	1.42E 02	3.58E 01	1.33E 01	2.20E-06	0.0
PM149	0.0	9.57E 04	8.23E 00	5.64E-08	3.87E-16	0.0	0.0
PM151	0.0	6.22E 04	1.14E-03	0.0	0.0	0.0	0.0
SM151	0.0	4.98E 02	5.00E 02	5.00E 02	4.99E 02	4.89E 02	3.94E 02
EU152	0.0	2.52E-01	2.51E-01	2.48E-01	2.46E-01	2.12E-01	4.45E-02
SM153	0.0	2.31E 04	5.72E-01	3.42E-10	0.0	0.0	0.0
GD153	0.0	2.19E-03	2.01E-03	1.69E-03	1.42E-03	9.47E-05	5.11E-17
EU154	0.0	2.04E 01	2.03E 01	2.02E 01	2.00E 01	1.79E 01	5.56E 00
EU155	0.0	3.52E 03	3.41E 03	3.20E 03	3.00E 03	1.11E 03	3.60E-02
EU156	0.0	7.05E 03	1.81E 03	1.13E 02	7.07E 00	0.0	0.0
TB160	0.0	5.99E 01	4.49E 01	2.52E 01	1.42E 01	1.59E-03	0.0
TB161	0.0	8.38E 02	4.12E 01	9.92E-02	2.39E-04	0.0	0.0
GD162	0.0	1.28E 02	1.21E 02	1.08E 02	9.61E 01	1.60E 01	1.19E-07
TB162M	0.0	1.28E 02	1.21E 02	1.08E 02	9.51E 01	1.60E 01	1.19E-07
DY166	0.0	1.44E 01	3.15E-02	1.53E-07	7.44E-13	0.0	0.0
HO166	0.0	1.49E 01	4.70E-02	2.29E-07	1.11E-12	0.0	0.0
SUBTOT	0.0	6.28E 06	2.08E 06	1.06E 06	6.79E 05	8.48E 04	1.25E 04
TOTALS	0.0	3.15E 07	2.08E 06	1.06E 06	6.79E 05	8.48E 04	1.25E 04

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
KR 85	0.0	1.34E 00	1.33E 00	1.32E 00	1.31E 00	1.10E 00	1.94E-01
RB 86	0.0	3.62E-01	1.19E-01	1.28E-02	1.38E-03	7.80E-19	0.0
SR 89	0.0	4.94E 02	3.32E 02	1.49E 02	6.70E 01	2.24E-04	0.0
SR 90	0.0	5.46E 00	5.45E 00	5.43E 00	5.41E 00	5.07E 00	2.61E 00
Y 90	0.0	2.41E 01	2.39E 01	2.38E 01	2.37E 01	2.23E 01	1.14E 01
Y 91	0.0	7.12E 02	5.02E 02	2.47E 02	1.22E 02	1.76E-03	0.0
ZR 95	0.0	1.57E 03	1.14E 03	6.01E 02	3.17E 02	1.33E-02	0.0
NB 95M	0.0	8.35E 00	6.44E 00	3.40E 00	1.79E 00	7.49E-05	1.09E-16
NB 95	0.0	1.33E 03	1.28E 03	8.94E 02	5.38E 02	2.62E-02	0.0
MO 99	0.0	1.65E 03	9.60E-01	3.25E-07	1.10E-13	0.0	0.0
TC 99M	0.0	2.70E 02	1.73E-01	5.87E-08	1.99E-14	0.0	0.0
RU103	0.0	1.14E 03	6.77E 02	2.37E 02	8.28E 01	5.33E-06	0.0
RH103M	0.0	8.31E 01	4.92E 01	1.72E 01	6.02E 00	3.87E-07	0.0
RU106	0.0	5.04E 00	4.77E 00	4.26E 00	3.80E 00	6.36E-01	5.18E-09
RH106	0.0	8.22E 02	7.77E 02	6.94E 02	6.19E 02	1.04E 02	8.45E-07
AG110M	0.0	1.39E-01	1.28E-01	1.09E-01	9.23E-02	6.91E-03	1.27E-14
AG111	0.0	2.95E 01	1.85E 00	7.21E-03	2.82E-05	0.0	0.0
CD115M	0.0	4.10E-01	2.53E-01	9.61E-02	3.65E-02	8.71E-09	0.0
SN123M	0.0	1.40E-01	1.18E-01	8.48E-02	6.08E-02	3.20E-04	0.0
SB124	0.0	1.44E-01	1.02E-01	5.09E-02	2.55E-02	4.59E-07	0.0
SN125	0.0	2.62E 01	2.87E 00	3.44E-02	4.12E-04	0.0	0.0
SB125	0.0	3.45E 00	3.50E 00	3.37E 00	3.23E 00	1.66E 00	1.62E-03
TE125M	0.0	2.68E-01	2.93E-01	3.18E-01	3.24E-01	1.76E-01	1.72E-04
SB126	0.0	1.69E 01	3.20E 00	1.15E-01	4.62E-03	4.89E-04	4.89E-04
SB127	0.0	1.12E 02	5.34E-01	1.17E-05	2.58E-10	0.0	0.0
TE127M	0.0	1.74E 00	1.52E 00	1.04E 00	7.07E-01	1.73E-03	0.0
TE127	0.0	3.09E 01	4.57E 00	3.03E 00	2.07E 00	5.05E-03	4.41E-10
TE129M	0.0	6.01E 01	3.28E 01	9.66E 00	2.84E 00	1.20E-08	0.0
TE129	0.0	2.78E 02	3.85E 01	1.13E 01	3.34E 00	1.41E-08	0.0
I131	0.0	8.10E 02	6.29E 01	3.59E-01	2.05E-03	0.0	0.0
XE131M	0.0	1.69E 00	6.15E-01	2.42E-02	7.47E-04	0.0	0.0
TE132	0.0	3.80E 02	6.33E-01	1.80E-06	5.14E-12	0.0	0.0
I132	0.0	4.55E 03	7.66E 00	2.18E-05	6.22E-11	0.0	0.0
XE133	0.0	3.74E 02	8.73E 00	3.27E-03	1.22E-06	0.0	0.0
CS134	0.0	4.17E 00	4.05E 00	3.84E 00	3.63E 00	1.51E 00	1.67E-04
CS136	0.0	1.14E 02	2.31E 01	9.42E-01	3.84E-02	0.0	0.0
CS137	0.0	1.32E 01	1.32E 01	1.32E 01	1.31E 01	1.23E 01	6.62E 00
BA137M	0.0	3.00E 01	3.00E 01	2.98E 01	2.97E 01	2.80E 01	1.50E 01
BA140	0.0	1.08E 03	2.13E 02	8.26E 00	3.20E-01	0.0	0.0
LA140	0.0	5.38E 03	1.20E 03	4.67E 01	1.81E 00	1.81E-18	0.0
CE141	0.0	5.39E 02	3.38E 02	9.37E 01	2.60E 01	4.22E-08	0.0
PR143	0.0	5.44E 02	1.57E 02	7.56E 00	3.63E-01	0.0	0.0
CE144	0.0	1.20E 02	1.11E 02	9.60E 01	8.29E 01	8.24E 00	2.90E-10
PR144	0.0	1.04E 03	9.43E 02	8.14E 02	7.03E 02	6.99E 01	2.46E-09
ND147	0.0	4.95E 02	7.61E 01	1.80E 00	4.24E-02	0.0	0.0
PM147	0.0	2.03E 01	2.08E 01	2.01E 01	1.92E 01	9.68E 00	7.65E-03
PM148M	0.0	2.50E 01	1.52E 01	5.65E 00	2.10E 00	3.48E-07	0.0

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
PM148	0.0	1.89E 01	1.16E 00	2.93E-01	1.09E-01	1.80E-08	0.0
SM151	0.0	8.68E-01	8.72E-01	8.71E-01	8.69E-01	8.52E-01	6.87E-01
EU154	0.0	1.91E-01	1.90E-01	1.89E-01	1.88E-01	1.68E-01	5.20E-02
EU155	0.0	2.96E 00	2.87E 00	2.69E 00	2.53E 00	9.38E-01	3.03E-05
EU156	0.0	7.44E 01	1.91E 01	1.19E 00	7.46E-02	0.0	0.0
TB160	0.0	5.04E-01	3.78E-01	2.12E-01	1.19E-01	1.34E-05	0.0
GD162	0.0	4.35E-01	4.11E-01	3.66E-01	3.27E-01	5.43E-02	4.04E-10
TB162M	0.0	3.56E-01	8.08E-01	7.21E-01	6.44E-01	1.07E-01	7.96E-10
SUBTOT	0.0	2.45E 04	8.14E 03	4.06E 03	2.69E 03	2.66E 02	3.66E 01
TOTALS	0.0	3.31E 05	8.14E 03	4.06E 03	2.69E 03	2.67E 02	3.66E 01

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
Y 91	0.0	4.27E 00	3.01E 00	1.48E 00	7.31E-01	1.05E-05	0.0
ZR 95	0.0	1.29E 03	9.36E 02	4.94E 02	2.60E 02	1.09E-02	0.0
NB 95M	0.0	8.35E 00	6.44E 00	3.40E 00	1.79E 00	7.49E-05	1.09E-16
NB 95	0.0	1.26E 03	1.21E 03	8.45E 02	5.09E 02	2.48E-02	0.0
MO 99	0.0	4.78E 02	2.78E-01	9.44E-08	3.20E-14	0.0	0.0
RU103	0.0	1.01E 03	5.96E 02	2.08E 02	7.29E 01	4.69E-06	0.0
RH106	0.0	9.87E 01	9.32E 01	8.32E 01	7.43E 01	1.24E 01	1.01E-07
AG110M	0.0	1.34E-01	1.23E-01	1.04E-01	8.86E-02	6.64E-03	1.22E-14
SN125	0.0	2.38E 00	2.61E-01	3.13E-03	3.75E-05	0.0	0.0
SB125	0.0	1.90E 00	1.93E 00	1.85E 00	1.78E 00	9.14E-01	8.91E-04
SB126	0.0	9.80E 00	1.86E 00	6.70E-02	2.68E-03	2.84E-04	2.84E-04
SB127	0.0	5.66E 01	2.71E-01	5.95E-06	1.31E-10	0.0	0.0
TE129M	0.0	7.21E 00	3.94E 00	1.16E 00	3.41E-01	1.44E-09	0.0
TE129	0.0	4.17E 01	5.78E 00	1.70E 00	5.01E-01	2.11E-09	0.0
I131	0.0	4.65E 02	3.61E 01	2.06E-01	1.18E-03	0.0	0.0
TE132	0.0	1.44E 02	2.41E-01	6.86E-07	1.95E-12	0.0	0.0
I132	0.0	3.55E 03	5.97E 00	1.70E-05	4.85E-11	0.0	0.0
CS134	0.0	3.71E 00	3.61E 00	3.41E 00	3.23E 00	1.35E 00	1.49E-04
CS136	0.0	1.07E 02	2.17E 01	8.85E-01	3.61E-02	0.0	0.0
BA137M	0.0	3.00E 01	3.00E 01	2.98E 01	2.97E 01	2.80E 01	1.50E 01
BA140	0.0	5.08E 02	1.00E 02	3.88E 00	1.51E-01	0.0	0.0
LA140	0.0	4.14E 03	9.27E 02	3.60E 01	1.40E 00	1.40E-18	0.0
CE144	0.0	3.23E 01	3.00E 01	2.59E 01	2.24E 01	2.23E 00	7.83E-11
PR144	0.0	2.48E 01	2.26E 01	1.95E 01	1.69E 01	1.68E 00	5.91E-11
ND147	0.0	2.13E 01	3.27E 00	7.72E-02	1.82E-03	0.0	0.0
PM148M	0.0	2.27E 01	1.38E 01	5.14E 00	1.91E 00	3.17E-07	0.0
PM148	0.0	8.69E 00	5.34E-01	1.35E-01	4.99E-02	8.29E-09	0.0
EU154	0.0	1.60E-01	1.60E-01	1.59E-01	1.58E-01	1.41E-01	4.37E-02
EU156	0.0	5.65E 01	1.45E 01	9.07E-01	5.67E-02	0.0	0.0
TB160	0.0	4.14E-01	3.10E-01	1.74E-01	9.78E-02	1.10E-05	0.0
SUBTOT	0.0	1.34E 04	4.07E 03	1.77E 03	9.98E 02	4.68E 01	1.50E 01
TOTALS	0.0	1.11E 05	4.07E 03	1.77E 03	9.98E 02	4.68E 01	1.51E 01

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	1.82E-03	1.81E-03	1.79E-03	1.78E-03	1.54E-03	3.35E-04
ZN	0.0	4.14E-02	9.04E-07	4.30E-16	0.0	0.0	0.0
GA	0.0	3.33E 00	1.82E-05	8.69E-15	0.0	0.0	0.0
GE	0.0	3.55E 00	5.18E-20	0.0	0.0	0.0	0.0
AS	0.0	6.09E 01	6.42E-07	4.04E-18	0.0	0.0	0.0
SE	0.0	9.45E 02	5.62E-06	5.62E-06	5.62E-06	5.62E-06	5.62E-06
BR	0.0	1.03E 04	4.28E-07	2.32E-19	0.0	0.0	0.0
KR	0.0	1.10E 04	1.33E 00	1.32E 00	1.31E 00	1.10E 00	1.94E-01
RB	0.0	2.70E 04	1.19E-01	1.28E-02	1.38E-03	8.38E-10	8.38E-10
SR	0.0	1.70E 04	3.37E 02	1.54E 02	7.24E 01	5.07E 00	2.61E 00
Y	0.0	2.71E 04	5.25E 02	2.71E 02	1.46E 02	2.23E 01	1.14E 01
ZR	0.0	4.22E 03	1.14E 03	6.01E 02	3.17E 02	1.33E-02	1.50E-05
NB	0.0	2.10E 04	1.28E 03	8.98E 02	5.40E 02	2.63E-02	1.78E-05
MO	0.0	1.48E 04	9.60E-01	3.25E-07	1.10E-13	0.0	0.0
TC	0.0	2.11E 04	1.74E-01	8.01E-04	8.01E-04	8.01E-04	8.01E-04
RU	0.0	4.07E 03	6.82E 02	2.41E 02	8.66E 01	6.36E-01	5.18E-09
RH	0.0	3.60E 03	8.26E 02	7.11E 02	6.25E 02	1.04E 02	8.45E-07
PD	0.0	3.20E 02	1.81E-10	0.0	0.0	0.0	0.0
AG	0.0	2.63E 02	1.98E 00	1.22E-01	9.78E-02	7.32E-03	1.35E-14
CD	0.0	7.56E 01	2.54E-01	9.69E-02	3.73E-02	7.00E-04	1.84E-04
IN	0.0	2.24E 02	2.96E-04	2.94E-06	1.28E-06	2.58E-12	0.0
SN	0.0	4.50E 03	2.99E 00	1.22E-01	6.36E-02	1.52E-03	8.72E-04
SB	0.0	2.04E 04	7.34E 00	3.54E 00	3.26E 00	1.66E 00	2.14E-03
TE	0.0	1.58E 04	7.84E 01	2.54E 01	9.28E 00	1.83E-01	1.72E-04
I	0.0	3.22E 04	7.06E 01	3.59E-01	2.05E-03	1.45E-06	1.45E-06
XE	0.0	1.69E 04	9.35E 00	2.74E-02	7.48E-04	0.0	0.0
CS	0.0	2.81E 04	4.04E 01	1.79E 01	1.68E 01	1.39E 01	6.62E 00
BA	0.0	1.02E 04	2.43E 02	3.81E 01	3.00E 01	2.80E 01	1.50E 01
LA	0.0	2.00E 04	1.20E 03	4.67E 01	1.81E 00	1.81E-18	0.0
CE	0.0	7.54E 03	4.49E 02	1.90E 02	1.09E 02	8.24E 00	2.90E-10
PR	0.0	9.56E 03	1.10E 03	8.22E 02	7.04E 02	6.99E 01	2.46E-09
ND	0.0	1.44E 03	7.61E 01	1.80E 00	4.24E-02	0.0	0.0
PM	0.0	1.21E 03	3.72E 01	2.60E 01	2.14E 01	9.68E 00	7.65E-03
SM	0.0	1.30E 02	8.73E-01	8.71E-01	8.69E-01	8.52E-01	6.87E-01
EU	0.0	2.23E 02	2.22E 01	4.08E 00	2.79E 00	1.11E 00	5.26E-02
GD	0.0	1.64E 01	4.11E-01	3.66E-01	3.27E-01	5.43E-02	4.04E-10
TB	0.0	3.22E 00	1.25E 00	9.34E-01	7.63E-01	1.07E-01	7.96E-10
DY	0.0	9.89E-02	2.67E-05	1.30E-10	6.31E-16	0.0	0.0
HO	0.0	6.51E-02	2.06E-04	1.00E-09	4.87E-15	0.0	0.0
TOTALS	0.0	3.31E 05	8.14E 03	4.06E 03	2.69E 03	2.67E 02	3.66E 01

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
BR 81	0.0	2.04E 00	2.04E 00	2.04E 00	2.04E 00	2.04E 00	2.04E 00
SE 82	0.0	3.90E 00	3.90E 00	3.90E 00	3.90E 00	3.90E 00	3.90E 00
KR 83	0.0	1.45E 01	1.45E 01	1.45E 01	1.45E 01	1.45E 01	1.45E 01
KR 84	0.0	2.65E 01	2.65E 01	2.65E 01	2.65E 01	2.65E 01	2.65E 01
KR 85	0.0	6.56E 00	6.53E 00	6.46E 00	6.39E 00	5.41E 00	9.50E-01
RB 85	0.0	1.33E 01	1.34E 01	1.34E 01	1.35E 01	1.45E 01	1.89E 01
KR 86	0.0	3.48E 01	3.48E 01	3.48E 01	3.48E 01	3.48E 01	3.48E 01
RB 87	0.0	4.50E 01	4.50E 01	4.50E 01	4.50E 01	4.50E 01	4.50E 01
SR 88	0.0	6.50E 01	6.50E 01	6.50E 01	6.50E 01	6.50E 01	6.50E 01
SR 89	0.0	9.23E 00	6.19E 00	2.78E 00	1.25E 00	4.17E-06	0.0
Y 89	0.0	6.78E 01	7.08E 01	7.43E 01	7.56E 01	7.71E 01	7.71E 01
SR 90	0.0	8.83E 01	8.81E 01	8.78E 01	8.74E 01	8.20E 01	4.21E 01
ZR 90	0.0	2.24E 00	2.42E 00	2.78E 00	3.13E 00	8.54E 00	4.84E 01
Y 91	0.0	1.46E 01	1.03E 01	5.08E 00	2.50E 00	3.61E-05	0.0
ZR 91	0.0	9.47E 01	9.90E 01	1.04E 02	1.07E 02	1.09E 02	1.09E 02
ZR 92	0.0	1.30E 02	1.30E 02	1.30E 02	1.30E 02	1.30E 02	1.30E 02
ZR 93	0.0	1.51E 02	1.52E 02	1.52E 02	1.52E 02	1.52E 02	1.52E 02
ZR 94	0.0	1.67E 02	1.67E 02	1.67E 02	1.67E 02	1.67E 02	1.67E 02
ZR 95	0.0	3.05E 01	2.22E 01	1.17E 01	6.17E 00	2.58E-04	0.0
NB 95	0.0	1.57E 01	1.49E 01	1.03E 01	6.18E 00	3.00E-04	0.0
MO 95	0.0	1.46E 02	1.56E 02	1.71E 02	1.80E 02	1.93E 02	1.93E 02
ZR 96	0.0	1.91E 02	1.91E 02	1.91E 02	1.91E 02	1.91E 02	1.91E 02
MO 97	0.0	2.23E 02	2.23E 02	2.23E 02	2.23E 02	2.23E 02	2.23E 02
MO 98	0.0	2.13E 02	2.13E 02	2.13E 02	2.13E 02	2.13E 02	2.13E 02
TC 99	0.0	2.17E 02	2.19E 02	2.19E 02	2.19E 02	2.19E 02	2.19E 02
MO100	0.0	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02
RU100	0.0	2.22E 00	2.22E 00	2.22E 00	2.22E 00	2.22E 00	2.22E 00
RU101	0.0	2.20E 02	2.20E 02	2.20E 02	2.20E 02	2.20E 02	2.20E 02
RU102	0.0	2.35E 02	2.35E 02	2.35E 02	2.35E 02	2.35E 02	2.35E 02
RU103	0.0	2.23E 01	1.32E 01	4.62E 00	1.62E 00	1.04E-07	0.0
RH103	0.0	2.07E 02	2.16E 02	2.24E 02	2.27E 02	2.29E 02	2.29E 02
RU104	0.0	1.98E 02	1.98E 02	1.98E 02	1.98E 02	1.98E 02	1.98E 02
PD105	0.0	1.30E 02	1.30E 02	1.30E 02	1.30E 02	1.30E 02	1.30E 02
RU106	0.0	7.49E 01	7.07E 01	6.31E 01	5.64E 01	9.44E 00	7.69E-08
PD106	0.0	5.66E 01	6.07E 01	6.83E 01	7.50E 01	1.22E 02	1.31E 02
PD107	0.0	9.80E 01	9.80E 01	9.80E 01	9.80E 01	9.80E 01	9.80E 01
PD108	0.0	4.72E 01	4.72E 01	4.72E 01	4.72E 01	4.72E 01	4.72E 01
AG109	0.0	4.92E 01	4.93E 01	4.93E 01	4.93E 01	4.93E 01	4.93E 01
PD110	0.0	3.56E 00	3.56E 00	3.56E 00	3.56E 00	3.56E 00	3.56E 00
CD111	0.0	8.72E 00	8.94E 00	8.95E 00	8.95E 00	8.95E 00	8.95E 00
CD112	0.0	5.02E 00	5.04E 00	5.04E 00	5.04E 00	5.04E 00	5.04E 00
SB121	0.0	2.38E 00	2.39E 00	2.39E 00	2.39E 00	2.40E 00	2.43E 00
SB125	0.0	3.34E 00	3.36E 00	3.23E 00	3.09E 00	1.59E 00	1.55E-03
SN126	0.0	7.82E 00	7.82E 00	7.82E 00	7.82E 00	7.82E 00	7.82E 00
I127	0.0	1.56E 01	1.60E 01	1.63E 01	1.65E 01	1.69E 01	1.69E 01
TE128	0.0	2.78E 01	2.78E 01	2.78E 01	2.78E 01	2.78E 01	2.78E 01
I129	0.0	6.78E 01	6.89E 01	6.98E 01	7.00E 01	7.01E 01	7.01E 01

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TE130	0.0	9.10E 01	9.10E 01	9.10E 01	9.10E 01	9.10E 01	9.10E 01
XE131	0.0	1.56E 02	1.59E 02	1.59E 02	1.59E 02	1.59E 02	1.59E 02
XE132	0.0	2.28E 02	2.30E 02	2.30E 02	2.30E 02	2.30E 02	2.30E 02
CS133	0.0	2.78E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02
XE134	0.0	3.25E 02	3.25E 02	3.25E 02	3.25E 02	3.25E 02	3.25E 02
CS134	0.0	2.15E 00	2.10E 00	1.98E 00	1.87E 00	7.81E-01	8.65E-05
CS135	0.0	3.06E 02	3.06E 02	3.06E 02	3.06E 02	3.06E 02	3.06E 02
XE136	0.0	2.91E 02	2.91E 02	2.91E 02	2.91E 02	2.91E 02	2.91E 02
BA136	0.0	6.66E 00	6.88E 00	6.93E 00	6.94E 00	6.94E 00	6.94E 00
CS137	0.0	3.00E 02	3.00E 02	2.98E 02	2.97E 02	2.80E 02	1.50E 02
BA137	0.0	6.79E 00	7.36E 00	8.50E 00	9.63E 00	2.69E 01	1.57E 02
BA138	0.0	2.96E 02	2.96E 02	2.96E 02	2.96E 02	2.96E 02	2.96E 02
LA139	0.0	2.84E 02	2.84E 02	2.84E 02	2.84E 02	2.84E 02	2.84E 02
CE140	0.0	2.80E 02	2.89E 02	2.91E 02	2.91E 02	2.91E 02	2.91E 02
CE141	0.0	2.33E 01	1.24E 01	3.42E 00	9.49E-01	1.54E-09	0.0
PR141	0.0	2.64E 02	2.75E 02	2.84E 02	2.86E 02	2.87E 02	2.87E 02
CE142	0.0	2.68E 02	2.68E 02	2.68E 02	2.68E 02	2.68E 02	2.68E 02
PR143	0.0	9.39E 00	2.29E 00	1.10E-01	5.28E-03	0.0	0.0
ND143	0.0	2.56E 02	2.64E 02	2.67E 02	2.67E 02	2.67E 02	2.67E 02
CE144	0.0	1.03E 02	9.60E 01	8.29E 01	7.16E 01	7.12E 00	2.51E-10
ND144	0.0	1.16E 02	1.24E 02	1.37E 02	1.48E 02	2.12E 02	2.20E 02
ND145	0.0	2.07E 02	2.07E 02	2.07E 02	2.07E 02	2.07E 02	2.07E 02
ND146	0.0	1.88E 02	1.88E 02	1.88E 02	1.88E 02	1.88E 02	1.88E 02
PM147	0.0	1.17E 02	1.18E 02	1.13E 02	1.08E 02	5.46E 01	4.32E-02
SM147	0.0	3.34E 01	3.60E 01	4.10E 01	4.58E 01	9.96E 01	1.54E 02
ND148	0.0	1.14E 02	1.14E 02	1.14E 02	1.14E 02	1.14E 02	1.14E 02
SM148	0.0	5.28E 00	5.48E 00	5.63E 00	5.69E 00	5.73E 00	5.73E 00
SM149	0.0	8.25E 01	8.30E 01	8.30E 01	8.30E 01	8.30E 01	8.30E 01
ND150	0.0	6.83E 01	6.83E 01	6.83E 01	6.83E 01	6.83E 01	6.83E 01
SM150	0.0	2.74E 00	2.74E 00	2.74E 00	2.74E 00	2.74E 00	2.74E 00
SM151	0.0	5.33E 01	5.34E 01	5.34E 01	5.33E 01	5.22E 01	4.21E 01
SM152	0.0	4.24E 01	4.24E 01	4.24E 01	4.24E 01	4.24E 01	4.24E 01
EU153	0.0	2.03E 01	2.04E 01	2.04E 01	2.04E 01	2.04E 01	2.04E 01
SM154	0.0	1.42E 01	1.42E 01	1.42E 01	1.42E 01	1.42E 01	1.42E 01
EU155	0.0	1.02E 01	9.84E 00	9.24E 00	8.67E 00	3.22E 00	1.04E-04
GD155	0.0	3.47E 00	3.78E 00	4.38E 00	4.94E 00	1.04E 01	1.36E 01
GD156	0.0	7.32E 00	7.58E 00	7.67E 00	7.67E 00	7.67E 00	7.67E 00
GD157	0.0	4.24E 00	4.25E 00	4.25E 00	4.25E 00	4.25E 00	4.25E 00
GD158	0.0	8.98E 00	8.98E 00	8.98E 00	8.98E 00	8.98E 00	8.98E 00
TB159	0.0	4.66E 00	4.68E 00	4.68E 00	4.68E 00	4.68E 00	4.68E 00
SUBTOT	0.0	8.57E 03	8.60E 03	8.60E 03	8.60E 03	8.60E 03	8.59E 03
TOTALS	0.0	8.63E 03	8.63E 03	8.63E 03	8.63E 03	8.63E 03	8.63E 03

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	1.85E-02	1.84E-02	1.83E-02	1.81E-02	1.56E-02	3.41E-03
ZN	0.0	9.20E-05	2.01E-09	9.57E-19	0.0	0.0	0.0
GA	0.0	2.93E-05	8.74E-10	4.16E-19	0.0	0.0	0.0
GE	0.0	6.22E-02	6.23E-02	6.23E-02	6.23E-02	6.23E-02	6.23E-02
AS	0.0	1.82E-02	1.79E-02	1.79E-02	1.79E-02	1.79E-02	1.79E-02
SE	0.0	5.98E 00	5.98E 00	5.98E 00	5.98E 00	5.98E 00	5.98E 00
BR	0.0	2.04E 00	2.04E 00	2.04E 00	2.04E 00	2.04E 00	2.04E 00
KR	0.0	8.24E 01	8.24E 01	8.23E 01	8.22E 01	8.13E 01	7.68E 01
RB	0.0	5.83E 01	5.83E 01	5.84E 01	5.85E 01	5.95E 01	6.39E 01
SR	0.0	1.63E 02	1.59E 02	1.56E 02	1.54E 02	1.47E 02	1.07E 02
Y	0.0	8.27E 01	8.12E 01	7.94E 01	7.83E 01	7.71E 01	7.71E 01
ZR	0.0	7.67E 02	7.63E 02	7.58E 02	7.56E 02	7.57E 02	7.97E 02
NB	0.0	1.58E 01	1.49E 01	1.03E 01	6.19E 00	5.80E-04	2.17E-03
MO	0.0	8.22E 02	8.30E 02	8.45E 02	8.54E 02	8.67E 02	8.67E 02
TC	0.0	2.17E 02	2.19E 02	2.19E 02	2.19E 02	2.19E 02	2.19E 02
RU	0.0	7.52E 02	7.38E 02	7.22E 02	7.12E 02	6.64E 02	6.54E 02
RH	0.0	2.07E 02	2.16E 02	2.24E 02	2.27E 02	2.29E 02	2.29E 02
PD	0.0	3.35E 02	3.40E 02	3.47E 02	3.54E 02	4.01E 02	4.11E 02
AG	0.0	4.94E 01	4.93E 01	4.93E 01	4.93E 01	4.93E 01	4.93E 01
CD	0.0	1.81E 01	1.83E 01	1.83E 01	1.83E 01	1.83E 01	1.83E 01
IN	0.0	9.86E-01	9.92E-01	9.94E-01	9.96E-01	9.99E-01	1.02E 00
SN	0.0	1.56E 01	1.55E 01	1.54E 01	1.54E 01	1.54E 01	1.54E 01
SB	0.0	7.24E 00	6.98E 00	6.85E 00	6.72E 00	5.22E 00	3.67E 00
TE	0.0	1.26E 02	1.23E 02	1.22E 02	1.22E 02	1.23E 02	1.24E 02
I	0.0	8.76E 01	8.51E 01	8.60E 01	8.65E 01	8.70E 01	8.70E 01
XE	0.0	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03
CS	0.0	8.87E 02	8.91E 02	8.90E 02	8.88E 02	8.70E 02	7.39E 02
BA	0.0	3.19E 02	3.13E 02	3.12E 02	3.13E 02	3.32E 02	4.62E 02
LA	0.0	2.86E 02	2.85E 02	2.84E 02	2.84E 02	2.84E 02	2.84E 02
CE	0.0	6.76E 02	6.65E 02	6.45E 02	6.32E 02	5.66E 02	5.59E 02
PR	0.0	2.73E 02	2.77E 02	2.84E 02	2.86E 02	2.87E 02	2.87E 02
ND	0.0	9.53E 02	9.66E 02	9.80E 02	9.92E 02	1.06E 03	1.06E 03
PM	0.0	1.18E 02	1.18E 02	1.13E 02	1.08E 02	5.46E 01	4.32E-02
SM	0.0	2.34E 02	2.37E 02	2.42E 02	2.47E 02	3.00E 02	3.44E 02
EU	0.0	3.21E 01	3.17E 01	3.11E 01	3.06E 01	2.61E 01	3.24E 01
GD	0.0	2.48E 01	2.54E 01	2.60E 01	2.66E 01	3.20E 01	3.57E 01
TB	0.0	4.73E 00	4.71E 00	4.70E 00	4.69E 00	4.68E 00	4.68E 00
DY	0.0	1.43E 00	1.47E 00	1.52E 00	1.55E 00	1.73E 00	1.76E 00
HO	0.0	2.78E-02	2.78E-02	2.78E-02	2.78E-02	2.78E-02	2.78E-02
ER	0.0	1.69E-02	1.72E-02	1.72E-02	1.72E-02	1.72E-02	1.72E-02
TOTALS	0.0	8.63E 03	8.63E 03	8.63E 03	8.63E 03	8.63E 03	8.63E 03

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H 3	0.0	1.79E 02	1.79E 02	1.77E 02	1.75E 02	1.52E 02	3.31E 01
SE 79	0.0	3.61E-02	3.61E-02	3.61E-02	3.61E-02	3.61E-02	3.61E-02
KR 85	0.0	2.57E 03	2.55E 03	2.53E 03	2.50E 03	2.11E 03	3.71E 02
RB 86	0.0	3.56E 02	1.17E 02	1.26E 01	1.36E 00	7.68E-16	0.0
SR 89	0.0	2.61E 05	1.75E 05	7.85E 04	3.53E 04	1.18E-01	0.0
SR 90	0.0	1.25E 04	1.25E 04	1.24E 04	1.24E 04	1.16E 04	5.96E 03
Y 90	0.0	1.26E 04	1.25E 04	1.24E 04	1.24E 04	1.16E 04	5.96E 03
Y 91	0.0	3.57E 05	2.51E 05	1.24E 05	6.11E 04	8.81E-01	0.0
ZR 93	0.0	3.88E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01	3.89E-01
NB 93M	0.0	1.90E-02	2.06E-02	2.36E-02	2.67E-02	7.14E-02	3.09E-01
ZR 95	0.0	6.46E 05	4.69E 05	2.47E 05	1.31E 05	5.46E 00	0.0
NB 95M	0.0	1.29E 04	9.96E 03	5.25E 03	2.77E 03	1.16E-01	1.68E-13
NB 95	0.0	6.16E 05	5.85E 05	4.05E 05	2.43E 05	1.18E 01	0.0
MO 99	0.0	7.53E 05	4.39E 02	1.49E-04	5.04E-11	0.0	0.0
TC 99M	0.0	6.55E 05	4.19E 02	1.42E-04	4.82E-11	0.0	0.0
TC 99	0.0	3.70E 00	3.73E 00	3.73E 00	3.73E 00	3.73E 00	3.73E 00
RU103	0.0	7.15E 05	4.23E 05	1.48E 05	5.18E 04	3.33E-03	0.0
RH103M	0.0	7.15E 05	4.24E 05	1.48E 05	5.18E 04	3.34E-03	0.0
RH105	0.0	4.44E 05	4.08E-01	3.43E-13	0.0	0.0	0.0
RU106	0.0	2.51E 05	2.37E 05	2.12E 05	1.89E 05	3.17E 04	2.58E-04
RH106	0.0	2.51E 05	2.37E 05	2.12E 05	1.89E 05	3.17E 04	2.58E-04
AG110M	0.0	8.19E 01	7.54E 01	6.40E 01	5.43E 01	4.06E 00	7.48E-12
AG110	0.0	5.42E 03	9.80E 00	8.32E 00	7.06E 00	5.28E-01	9.72E-13
AG111	0.0	3.51E 04	2.20E 03	8.58E 00	3.35E-02	0.0	0.0
CD113M	0.0	6.37E 00	6.34E 00	6.29E 00	6.24E 00	5.49E 00	1.44E 00
IN114M	0.0	4.87E-02	3.21E-02	1.40E-02	6.08E-03	1.23E-08	0.0
IN114	0.0	7.99E-02	3.10E-02	1.35E-02	5.87E-03	1.18E-08	0.0
CD115M	0.0	1.66E 02	1.02E 02	3.89E 01	1.48E 01	3.52E-06	0.0
CD115	0.0	2.41E 03	2.15E-01	1.70E-09	1.34E-17	0.0	0.0
IN115M	0.0	2.41E 03	2.35E-01	1.86E-09	1.47E-17	0.0	0.0
SN117M	0.0	2.07E-01	4.68E-02	2.40E-03	1.23E-04	0.0	0.0
SN119M	0.0	9.15E 00	8.42E 00	7.13E 00	6.03E 00	4.38E-01	5.83E-13
SN121M	0.0	6.23E 00	6.22E 00	6.21E 00	6.20E 00	6.06E 00	4.74E 00
SB122	0.0	1.06E 02	6.33E-02	2.24E-08	7.95E-15	0.0	0.0
SN123M	0.0	1.26E 02	1.06E 02	7.62E 01	5.45E 01	2.88E-01	0.0
TE123M	0.0	9.91E-02	8.30E-02	5.81E-02	4.08E-02	1.50E-04	0.0
SB124	0.0	4.24E 01	3.00E 01	1.50E 01	7.49E 00	1.35E-04	0.0
SN125	0.0	1.14E 04	1.24E 03	1.49E 01	1.79E-01	0.0	0.0
SB125	0.0	3.54E 03	3.56E 03	3.42E 03	3.28E 03	1.69E 03	1.64E 00
TE125M	0.0	1.18E 03	1.24E 03	1.30E 03	1.31E 03	6.99E 02	6.81E-01
SN126	0.0	2.22E-01	2.22E-01	2.22E-01	2.22E-01	2.22E-01	2.22E-01
SB126M	0.0	2.16E 03	2.22E-01	2.22E-01	2.22E-01	2.22E-01	2.22E-01
SB126	0.0	2.25E 03	4.27E 02	1.56E 01	7.70E-01	2.20E-01	2.20E-01
SB127	0.0	5.83E 04	2.78E 02	6.12E-03	1.34E-07	0.0	0.0
TE127M	0.0	1.04E 04	9.03E 03	6.17E 03	4.21E 03	1.03E 01	0.0
TE127	0.0	5.58E 04	9.17E 03	6.10E 03	4.16E 03	1.02E 01	8.87E-07
TE129M	0.0	6.93E 04	3.78E 04	1.11E 04	3.27E 03	1.38E-05	0.0

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TE129	0.0	1.72E 05	2.42E 04	7.13E 03	2.10E 03	8.84E-06	0.0
I129	0.0	1.11E-02	1.12E-02	1.14E-02	1.14E-02	1.14E-02	1.14E-02
I131	0.0	4.17E 05	3.24E 04	1.85E 02	1.05E 00	0.0	0.0
XE131M	0.0	3.98E 03	1.39E 03	5.42E 01	1.67E 00	0.0	0.0
TE132	0.0	5.88E 05	9.80E 02	2.79E-03	7.96E-09	0.0	0.0
I132	0.0	5.96E 05	1.01E 03	2.88E-03	8.20E-09	0.0	0.0
XE133M	0.0	1.68E 04	2.71E 00	2.76E-08	2.81E-16	0.0	0.0
XE133	0.0	7.46E 05	1.72E 04	6.44E 00	2.41E-03	0.0	0.0
CS134	0.0	2.80E 03	2.73E 03	2.58E 03	2.44E 03	1.02E 03	1.13E-01
CS135	0.0	2.70E-01	2.70E-01	2.70E-01	2.70E-01	2.70E-01	2.70E-01
CS136	0.0	1.99E 04	4.02E 03	1.64E 02	6.69E 00	0.0	0.0
CS137	0.0	2.61E 04	2.61E 04	2.60E 04	2.59E 04	2.44E 04	1.31E 04
BA137M	0.0	2.44E 04	2.44E 04	2.43E 04	2.42E 04	2.28E 04	1.22E 04
BA140	0.0	6.57E 05	1.29E 05	5.02E 03	1.95E 02	0.0	0.0
LA140	0.0	6.71E 05	1.49E 05	5.78E 03	2.24E 02	2.24E-16	0.0
CE141	0.0	6.68E 05	3.53E 05	9.79E 04	2.71E 04	4.41E-05	0.0
CE143	0.0	6.26E 05	1.71E-01	1.25E-14	0.0	0.0	0.0
PR143	0.0	6.26E 05	1.53E 05	7.33E 03	3.52E 02	0.0	0.0
CE144	0.0	3.30E 05	3.07E 05	2.65E 05	2.29E 05	2.27E 04	8.00E-07
PR144	0.0	3.39E 05	3.07E 05	2.65E 05	2.29E 05	2.27E 04	8.00E-07
ND147	0.0	3.35E 05	5.15E 04	1.22E 03	2.87E 01	0.0	0.0
PM147	0.0	1.08E 05	1.09E 05	1.05E 05	1.01E 05	5.07E 04	4.01E 01
PM148M	0.0	8.24E 03	5.02E 03	1.86E 03	6.93E 02	1.15E-04	0.0
PM148	0.0	9.12E 03	5.83E 02	1.50E 02	5.57E 01	9.23E-06	0.0
PM149	0.0	1.86E 05	1.60E 01	1.10E-07	7.53E-16	0.0	0.0
SM151	0.0	1.45E 03	1.46E 03	1.45E 03	1.45E 03	1.42E 03	1.15E 03
EU152	0.0	2.62E 00	2.61E 00	2.59E 00	2.56E 00	2.21E 00	4.64E-01
SM153	0.0	4.72E 04	1.16E 00	6.97E-10	0.0	0.0	0.0
GD153	0.0	5.41E-02	4.97E-02	4.18E-02	3.52E-02	2.34E-03	1.26E-15
EU154	0.0	1.35E 02	1.35E 02	1.34E 02	1.33E 02	1.19E 02	3.69E 01
EU155	0.0	1.29E 04	1.25E 04	1.18E 04	1.11E 04	4.10E 03	1.32E-01
EU156	0.0	1.94E 04	4.97E 03	3.11E 02	1.94E 01	2.01E-18	0.0
TB160	0.0	5.43E 02	4.07E 02	2.29E 02	1.28E 02	1.44E-02	0.0
TB161	0.0	2.66E 03	1.31E 02	3.16E-01	7.61E-04	0.0	0.0
GD162	0.0	5.80E 02	5.48E 02	4.89E 02	4.36E 02	7.24E 01	5.39E-07
TB162M	0.0	5.80E 02	5.48E 02	4.89E 02	4.36E 02	7.24E 01	5.39E-07
DY166	0.0	4.55E 01	9.96E-02	4.84E-07	2.36E-12	0.0	0.0
HO166	0.0	4.93E 01	1.49E-01	7.23E-07	3.52E-12	0.0	0.0
SUBTOT	0.0	1.32E 07	4.62E 06	2.47E 06	1.65E 06	2.41E 05	3.88E 04
TOTALS	0.0	6.41E 07	4.62E 06	2.47E 06	1.65E 06	2.41E 05	3.88E 04

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
KR 85	0.0	4.12E 00	4.10E 00	4.06E 00	4.01E 00	3.40E 00	5.97E-01
RB 86	0.0	1.68E 00	5.50E-01	5.93E-02	6.38E-03	3.61E-18	0.0
SR 89	0.0	9.38E 02	6.29E 02	2.83E 02	1.27E 02	4.24E-04	0.0
SR 90	0.0	1.63E 01	1.62E 01	1.62E 01	1.61E 01	1.51E 01	7.77E 00
Y 90	0.0	7.23E 01	7.13E 01	7.10E 01	7.07E 01	6.63E 01	3.41E 01
Y 91	0.0	1.36E 03	9.56E 02	4.72E 02	2.32E 02	3.35E-03	0.0
ZR 95	0.0	3.38E 03	2.46E 03	1.30E 03	6.83E 02	2.86E-02	0.0
NB 95M	0.0	1.80E 01	1.39E 01	7.32E 00	3.86E 00	1.61E-04	2.35E-16
NB 95	0.0	2.95E 03	2.81E 03	1.94E 03	1.16E 03	5.65E-02	0.0
MO 99	0.0	3.38E 03	1.97E 00	6.68E-07	2.26E-13	0.0	0.0
TC 99M	0.0	5.55E 02	3.55E-01	1.20E-07	4.08E-14	0.0	0.0
RU103	0.0	2.34E 03	1.38E 03	4.83E 02	1.69E 02	1.09E-05	0.0
RH103M	0.0	1.70E 02	1.00E 02	3.51E 01	1.23E 01	7.91E-07	0.0
RU106	0.0	1.49E 01	1.41E 01	1.26E 01	1.12E 01	1.88E 00	1.53E-08
RH106	0.0	2.43E 03	2.29E 03	2.05E 03	1.83E 03	3.06E 02	2.49E-06
AG110M	0.0	1.32E 00	1.22E 00	1.03E 00	8.75E-01	6.55E-02	1.21E-13
AG111	0.0	8.41E 01	5.26E 00	2.05E-02	8.02E-05	0.0	0.0
CD115M	0.0	6.75E-01	4.16E-01	1.58E-01	6.02E-02	1.43E-08	0.0
SN123M	0.0	4.28E-01	3.62E-01	2.60E-01	1.86E-01	9.81E-04	0.0
SB124	0.0	5.73E-01	4.05E-01	2.03E-01	1.01E-01	1.83E-06	0.0
SN125	0.0	6.88E 01	7.53E 00	9.03E-02	1.08E-03	0.0	0.0
SB125	0.0	1.19E 01	1.19E 01	1.15E 01	1.10E 01	5.66E 00	5.52E-03
TE125M	0.0	1.01E 00	1.07E 00	1.12E 00	1.12E 00	6.01E-01	5.85E-04
SB126	0.0	2.47E 01	4.69E 00	1.71E-01	8.45E-03	2.41E-03	2.41E-03
SB127	0.0	3.20E 02	1.53E 00	3.36E-05	7.39E-10	0.0	0.0
TE127M	0.0	5.76E 00	4.98E 00	3.40E 00	2.32E 00	5.67E-03	0.0
TE127	0.0	9.09E 01	1.50E 01	9.94E 00	6.79E 00	1.66E-02	1.45E-09
TE129M	0.0	1.37E 02	7.49E 01	2.20E 01	6.48E 00	2.73E-08	0.0
TE129	0.0	6.25E 02	8.79E 01	2.59E 01	7.61E 00	3.21E-08	0.0
I131	0.0	1.72E 03	1.33E 02	7.61E-01	4.34E-03	0.0	0.0
XE131M	0.0	3.87E 00	1.35E 00	5.27E-02	1.63E-03	0.0	0.0
TE132	0.0	8.02E 02	1.34E 00	3.81E-06	1.08E-11	0.0	0.0
I132	0.0	9.53E 03	1.62E 01	4.61E-05	1.31E-10	0.0	0.0
XE133	0.0	8.04E 02	1.86E 01	6.95E-03	2.60E-06	0.0	0.0
CS134	0.0	2.94E 01	2.86E 01	2.71E 01	2.56E 01	1.07E 01	1.18E-03
CS136	0.0	3.08E 02	6.22E 01	2.54E 00	1.04E-01	0.0	0.0
CS137	0.0	4.23E 01	4.22E 01	4.20E 01	4.19E 01	3.94E 01	2.11E 01
BA137M	0.0	9.58E 01	9.56E 01	9.53E 01	9.49E 01	8.94E 01	4.79E 01
BA140	0.0	2.22E 03	4.37E 02	1.69E 01	6.57E-01	0.0	0.0
LA140	0.0	1.11E 04	2.47E 03	9.59E 01	3.72E 00	3.72E-18	0.0
CE141	0.0	1.31E 03	6.96E 02	1.93E 02	5.34E 01	8.67E-08	0.0
PR143	0.0	1.36E 03	3.31E 02	1.59E 01	7.64E-01	0.0	0.0
CE144	0.0	2.89E 02	2.69E 02	2.32E 02	2.01E 02	1.99E 01	7.02E-10
PR144	0.0	2.52E 03	2.28E 03	1.97E 03	1.70E 03	1.69E 02	5.96E-09
ND147	0.0	9.38E 02	1.44E 02	3.40E 00	8.03E-02	0.0	0.0
PM147	0.0	5.58E 01	5.63E 01	5.42E 01	5.19E 01	2.62E 01	2.07E-02
PM148M	0.0	1.05E 02	6.37E 01	2.37E 01	8.80E 00	1.46E-06	0.0

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
PM148	0.0	7.46E 01	4.77E 00	1.23E 00	4.55E-01	7.55E-08	0.0
SM151	0.0	2.53E 00	2.54E 00	2.53E 00	2.53E 00	2.48E 00	2.00E 00
EU154	0.0	1.27E 00	1.26E 00	1.25E 00	1.25E 00	1.11E 00	3.45E-01
EU155	0.0	1.09E 01	1.06E 01	9.91E 00	9.31E 00	3.45E 00	1.12E-04
EU156	0.0	2.05E 02	5.25E 01	3.28E 00	2.05E-01	2.12E-20	0.0
TB160	0.0	4.57E 00	3.43E 00	1.93E 00	1.08E 00	1.21E-04	0.0
TB161	0.0	4.34E 00	2.13E-01	5.14E-04	1.24E-06	0.0	0.0
GD162	0.0	1.97E 00	1.86E 00	1.66E 00	1.48E 00	2.46E-01	1.83E-09
TB162M	0.0	3.88E 00	3.67E 00	3.27E 00	2.92E 00	4.85E-01	3.61E-09
SUBTOT	0.0	5.26E 04	1.82E 04	9.54E 03	6.56E 03	7.62E 02	1.14E 02
TOTALS	0.0	6.66E 05	1.82E 04	9.54E 03	6.56E 03	7.62E 02	1.14E 02

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
Y 91	0.0	8.15E 00	5.74E 00	2.83E 00	1.39E 00	2.01E-05	0.0
ZR 95	0.0	2.78E 03	2.02E 03	1.06E 03	5.61E 02	2.35E-02	0.0
NB 95M	0.0	1.80E 01	1.39E 01	7.32E 00	3.86E 00	1.61E-04	2.35E-16
NB 95	0.0	2.79E 03	2.65E 03	1.83E 03	1.10E 03	5.34E-02	0.0
MO 99	0.0	9.81E 02	5.72E-01	1.94E-07	6.57E-14	0.0	0.0
RU103	0.0	2.06E 03	1.22E 03	4.25E 02	1.49E 02	9.58E-06	0.0
RH106	0.0	2.91E 02	2.75E 02	2.46E 02	2.19E 02	3.68E 01	2.99E-07
AG110M	0.0	1.27E 00	1.17E 00	9.90E-01	8.40E-01	6.29E-02	1.16E-13
SB124	0.0	4.70E-01	3.32E-01	1.66E-01	8.30E-02	1.50E-06	0.0
SN125	0.0	6.26E 00	6.85E-01	8.21E-03	9.84E-05	0.0	0.0
SB125	0.0	6.52E 00	6.56E 00	6.31E 00	6.05E 00	3.11E 00	3.03E-03
SB126	0.0	1.43E 01	2.72E 00	9.89E-02	4.90E-03	1.40E-03	1.40E-03
SB127	0.0	1.62E 02	7.76E-01	1.70E-05	3.74E-10	0.0	0.0
TE127	0.0	1.36E 00	2.24E-01	1.49E-01	1.02E-01	2.49E-04	2.17E-11
TE129M	0.0	1.65E 01	8.98E 00	2.64E 00	7.78E-01	3.28E-09	0.0
TE129	0.0	9.38E 01	1.32E 01	3.88E 00	1.14E 00	4.81E-09	0.0
I131	0.0	9.85E 02	7.65E 01	4.37E-01	2.49E-03	0.0	0.0
TE132	0.0	3.05E 02	5.08E-01	1.45E-06	4.12E-12	0.0	0.0
I132	0.0	7.44E 03	1.26E 01	3.59E-05	1.02E-10	0.0	0.0
CS134	0.0	2.62E 01	2.55E 01	2.41E 01	2.28E 01	9.49E 00	1.05E-03
CS136	0.0	2.90E 02	5.85E 01	2.39E 00	9.73E-02	0.0	0.0
BA137M	0.0	9.58E 01	9.56E 01	9.53E 01	9.49E 01	8.94E 01	4.79E 01
BA140	0.0	1.04E 03	2.05E 02	7.96E 00	3.09E-01	0.0	0.0
LA140	0.0	8.57E 03	1.90E 03	7.39E 01	2.87E 00	2.86E-18	0.0
CE144	0.0	7.82E 01	7.26E 01	6.27E 01	5.42E 01	5.38E 00	1.90E-10
PR144	0.0	6.06E 01	5.47E 01	4.73E 01	4.08E 01	4.06E 00	1.43E-10
ND147	0.0	4.03E 01	6.20E 00	1.46E-01	3.45E-03	0.0	0.0
PM148M	0.0	9.52E 01	5.80E 01	2.15E 01	8.00E 00	1.33E-06	0.0
PM148	0.0	3.43E 01	2.19E 00	5.64E-01	2.09E-01	3.47E-08	0.0
EU154	0.0	1.06E 00	1.06E 00	1.05E 00	1.05E 00	9.35E-01	2.90E-01
EU156	0.0	1.56E 02	3.99E 01	2.49E 00	1.56E-01	1.61E-20	0.0
TB160	0.0	3.75E 00	2.81E 00	1.58E 00	8.87E-01	9.95E-05	0.0
SUBTOT	0.0	2.84E 04	8.83E 03	3.94E 03	2.27E 03	1.49E 02	4.82E 01
TOTALS	0.0	2.26E 05	8.83E 03	3.94E 03	2.27E 03	1.49E 02	4.82E 01

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	6.38E-03	6.35E-03	6.29E-03	6.24E-03	5.39E-03	1.18E-03
ZN	0.0	1.26E-01	2.74E-06	1.31E-15	0.0	0.0	0.0
GA	0.0	5.58E 00	5.54E-05	2.64E-14	0.0	0.0	0.0
GE	0.0	4.94E 00	7.18E-20	0.0	0.0	0.0	0.0
AS	0.0	8.54E 01	8.90E-07	5.60E-18	0.0	0.0	0.0
SE	0.0	1.73E 03	1.37E-05	1.37E-05	1.37E-05	1.37E-05	1.37E-05
BR	0.0	1.80E 04	1.48E-06	8.02E-19	0.0	0.0	0.0
KR	0.0	2.14E 04	4.10E 00	4.06E 00	4.01E 00	3.40E 00	5.97E-01
RB	0.0	5.46E 04	5.50E-01	5.93E-02	6.38E-03	2.41E-09	2.41E-09
SR	0.0	3.42E 04	6.45E 02	2.99E 02	1.43E 02	1.51E 01	7.77E 00
Y	0.0	5.28E 04	1.03E 03	5.43E 02	3.03E 02	6.63E 01	3.41E 01
ZR	0.0	8.56E 03	2.46E 03	1.30E 03	6.83E 02	2.86E-02	4.61E-05
NB	0.0	4.29E 04	2.82E 03	1.95E 03	1.17E 03	5.67E-02	5.49E-05
MO	0.0	2.91E 04	1.97E 00	6.68E-07	2.26E-13	0.0	0.0
TC	0.0	4.24E 04	3.58E-01	2.52E-03	2.52E-03	2.52E-03	2.52E-03
RU	0.0	8.96E 03	1.40E 03	4.96E 02	1.80E 02	1.88E 00	1.53E-08
RH	0.0	9.40E 03	2.39E 03	2.08E 03	1.84E 03	3.06E 02	2.49E-06
PD	0.0	8.62E 02	5.06E-10	0.0	0.0	0.0	0.0
AG	0.0	6.62E 02	6.54E 00	1.11E 00	9.26E-01	6.94E-02	1.28E-13
CD	0.0	1.22E 02	4.26E-01	1.67E-01	6.84E-02	7.26E-03	1.91E-03
IN	0.0	3.36E 02	6.58E-04	8.34E-05	3.63E-05	7.31E-11	0.0
SN	0.0	6.82E 03	7.91E 00	3.60E-01	1.97E-01	7.69E-03	5.09E-03
SB	0.0	4.13E 04	1.86E 01	1.18E 01	1.11E 01	5.66E 00	8.05E-03
TE	0.0	3.17E 04	1.85E 02	6.24E 01	2.43E 01	6.23E-01	5.85E-04
I	0.0	6.48E 04	1.50E 02	7.61E-01	4.34E-03	4.95E-06	4.95E-06
XE	0.0	3.61E 04	1.99E 01	5.96E-02	1.63E-03	0.0	0.0
CS	0.0	5.95E 04	1.33E 02	7.16E 01	6.76E 01	5.01E 01	2.11E 01
BA	0.0	2.12E 04	5.32E 02	1.12E 02	9.56E 01	8.94E 01	4.79E 01
LA	0.0	4.03E 04	2.47E 03	9.59E 01	3.72E 00	3.72E-18	0.0
CE	0.0	1.41E 04	9.65E 02	4.25E 02	2.54E 02	1.99E 01	7.02E-10
PR	0.0	1.81E 04	2.61E 03	1.99E 03	1.70E 03	1.69E 02	5.96E-09
ND	0.0	2.75E 03	1.44E 02	3.40E 00	8.03E-02	0.0	0.0
PM	0.0	2.31E 03	1.25E 02	7.91E 01	6.12E 01	2.62E 01	2.07E-02
SM	0.0	3.01E 02	2.54E 00	2.53E 00	2.53E 00	2.48E 00	2.00E 00
EU	0.0	6.66E 02	6.43E 01	1.45E 01	1.08E 01	4.59E 00	3.51E-01
GD	0.0	5.31E 01	1.86E 00	1.66E 00	1.48E 00	2.46E-01	1.83E-09
TB	0.0	1.42E 01	7.31E 00	5.20E 00	4.00E 00	4.85E-01	3.61E-09
DY	0.0	3.16E-01	8.45E-05	4.11E-10	2.00E-15	0.0	0.0
HO	0.0	2.16E-01	6.51E-04	3.17E-09	1.54E-14	0.0	0.0
TOTALS	0.0	6.66E 05	1.82E 04	9.54E 03	6.55E 03	7.62E 02	1.14E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
TE125	5.49E 00	6.11E 00	7.36E 00	8.58E 00	2.26E 01	3.75E 01
SN126	9.87E 01	9.87E 01	9.87E 01	9.87E 01	9.87E 01	9.86E 01
TE127M	1.23E 01	1.06E 01	7.23E 00	4.93E 00	1.20E-02	0.0
I127	1.82E 02	1.86E 02	1.89E 02	1.91E 02	1.96E 02	1.96E 02
TE128	2.60E 02	2.60E 02	2.60E 02	2.60E 02	2.60E 02	2.60E 02
XE128	6.19E 00	6.19E 00	6.19E 00	6.19E 00	6.19E 00	6.19E 00
TE129M	1.77E 01	9.67E 00	2.85E 00	8.37E-01	3.53E-09	0.0
I129	5.11E 02	5.19E 02	5.26E 02	5.28E 02	5.28E 02	5.28E 02
TE130	5.70E 02	5.70E 02	5.70E 02	5.70E 02	5.70E 02	5.70E 02
XE130	1.63E 01	1.64E 01	1.64E 01	1.64E 01	1.64E 01	1.64E 01
XE131	1.05E 03	1.07E 03	1.07E 03	1.07E 03	1.07E 03	1.07E 03
XE132	1.52E 03	1.53E 03	1.53E 03	1.53E 03	1.53E 03	1.53E 03
CS133	1.92E 03	1.96E 03	1.96E 03	1.96E 03	1.96E 03	1.96E 03
XE134	2.15E 03	2.15E 03	2.15E 03	2.15E 03	2.15E 03	2.15E 03
CS134	3.83E 01	3.73E 01	3.53E 01	3.33E 01	1.39E 01	1.54E-03
BA134	6.59E 00	7.64E 00	9.65E 00	1.16E 01	3.10E 01	4.49E 01
CS135	2.09E 03	2.09E 03	2.09E 03	2.09E 03	2.09E 03	2.09E 03
XE136	1.91E 03	1.91E 03	1.91E 03	1.91E 03	1.91E 03	1.91E 03
BA136	5.85E 01	6.10E 01	6.16E 01	6.17E 01	6.17E 01	6.17E 01
CS137	1.97E 03	1.96E 03	1.96E 03	1.95E 03	1.84E 03	9.84E 02
BA137	3.36E 01	3.73E 01	4.47E 01	5.21E 01	1.65E 02	1.02E 03
BA138	1.90E 03	1.90E 03	1.90E 03	1.90E 03	1.90E 03	1.90E 03
LA139	1.78E 03	1.78E 03	1.78E 03	1.78E 03	1.78E 03	1.78E 03
BA140	5.68E 01	1.12E 01	4.34E-01	1.69E-02	0.0	0.0
CE140	1.91E 03	1.96E 03	1.97E 03	1.98E 03	1.98E 03	1.98E 03
CE141	1.53E 02	8.12E 01	2.25E 01	6.23E 00	1.01E-08	0.0
PR141	1.61E 03	1.68E 03	1.74E 03	1.75E 03	1.76E 03	1.76E 03
CE142	1.56E 03	1.56E 03	1.56E 03	1.56E 03	1.56E 03	1.56E 03
ND142	7.89E 00	7.93E 00	7.93E 00	7.93E 00	7.93E 00	7.93E 00
PR143	6.22E 01	1.52E 01	7.28E-01	3.50E-02	5.71E-23	0.0
ND143	1.55E 03	1.60E 03	1.62E 03	1.62E 03	1.62E 03	1.62E 03
CE144	6.74E 02	6.26E 02	5.41E 02	4.67E 02	4.64E 01	1.64E-09
ND144	6.38E 02	6.86E 02	7.71E 02	8.45E 02	1.27E 03	1.31E 03
ND145	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03
ND146	9.38E 02	9.38E 02	9.38E 02	9.38E 02	9.38E 02	9.38E 02
PM147	5.69E 02	5.76E 02	5.55E 02	5.31E 02	2.68E 02	2.11E-01
SM147	1.13E 02	1.25E 02	1.50E 02	1.74E 02	4.37E 02	7.04E 02
ND148	5.83E 02	5.83E 02	5.83E 02	5.83E 02	5.83E 02	5.83E 02
SM148	5.32E 01	5.60E 01	5.81E 01	5.89E 01	5.94E 01	5.94E 01
SM149	4.29E 02	4.32E 02	4.32E 02	4.32E 02	4.32E 02	4.32E 02
ND150	3.44E 02	3.44E 02	3.44E 02	3.44E 02	3.44E 02	3.44E 02
SM150	3.41E 01	3.41E 01	3.41E 01	3.41E 01	3.41E 01	3.41E 01
SM151	2.60E 02	2.61E 02	2.61E 02	2.60E 02	2.55E 02	2.06E 02
SM152	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02
EU153	1.15E 02	1.15E 02	1.15E 02	1.15E 02	1.15E 02	1.15E 02
SM154	9.48E 01	9.48E 01	9.48E 01	9.48E 01	9.48E 01	9.48E 01
EU154	1.11E 01	1.10E 01	1.10E 01	1.09E 01	9.73E 00	3.02E 00
EU155	1.05E 02	1.01E 02	9.52E 01	8.94E 01	3.32E 01	1.07E-03
SE 82	4.59E 00	4.59E 00	4.59E 00	4.59E 00	4.59E 00	4.59E 00
KR 83	1.03E 02	1.03E 02	1.03E 02	1.03E 02	1.03E 02	1.03E 02
KR 84	1.79E 02	1.79E 02	1.79E 02	1.79E 02	1.79E 02	1.79E 02
KR 85	4.09E 01	4.07E 01	4.03E 01	3.99E 01	3.37E 01	5.93E 00
RB 85	6.67E 01	6.70E 01	6.74E 01	6.78E 01	7.39E 01	1.02E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
KR 86	1.62E 02	1.62E 02	1.62E 02	1.62E 02	1.62E 02	1.62E 02
RB 87	2.00E 02	2.00E 02	2.00E 02	2.00E 02	2.00E 02	2.00E 02
SR 88	3.08E 02	3.08E 02	3.08E 02	3.08E 02	3.08E 02	3.08E 02
SR 89	5.19E 01	3.48E 01	1.56E 01	7.03E 00	2.35E-05	0.0
Y 89	3.19E 02	3.36E 02	3.56E 02	3.64E 02	3.71E 02	3.71E 02
SR 90	4.71E 02	4.71E 02	4.69E 02	4.67E 02	4.38E 02	2.25E 02
ZR 90	8.86E 00	9.82E 00	1.17E 01	1.36E 01	4.25E 01	2.55E 02
Y 91	8.26E 01	5.82E 01	2.87E 01	1.41E 01	2.04E-04	0.0
ZR 91	4.41E 02	4.65E 02	4.95E 02	5.10E 02	5.24E 02	5.24E 02
ZR 92	6.87E 02	6.88E 02	6.88E 02	6.88E 02	6.88E 02	6.88E 02
ZR 93	8.56E 02	8.57E 02	8.57E 02	8.57E 02	8.57E 02	8.57E 02
ZR 94	9.76E 02	9.76E 02	9.76E 02	9.76E 02	9.76E 02	9.76E 02
ZR 95	2.16E 02	1.57E 02	8.26E 01	4.35E 01	1.82E-03	0.0
NB 95	1.15E 02	1.07E 02	7.34E 01	4.38E 01	2.12E-03	0.0
MO 95	8.95E 02	9.62E 02	1.07E 03	1.14E 03	1.23E 03	1.23E 03
ZR 96	1.14E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03
MO 96	2.15E 01	2.16E 01	2.16E 01	2.16E 01	2.16E 01	2.16E 01
MO 97	1.55E 03	1.55E 03	1.55E 03	1.55E 03	1.55E 03	1.55E 03
MO 98	1.34E 03	1.34E 03	1.34E 03	1.34E 03	1.34E 03	1.34E 03
TC 99	1.34E 03	1.36E 03	1.36E 03	1.36E 03	1.36E 03	1.36E 03
MO100	1.58E 03	1.58E 03	1.58E 03	1.58E 03	1.58E 03	1.58E 03
RU100	3.37E 01	3.37E 01	3.37E 01	3.37E 01	3.37E 01	3.37E 01
RU101	1.29E 03	1.29E 03	1.29E 03	1.29E 03	1.29E 03	1.29E 03
RU102	1.46E 03	1.46E 03	1.46E 03	1.46E 03	1.46E 03	1.46E 03
RU103	1.45E 02	8.60E 01	3.01E 01	1.05E 01	6.78E-07	0.0
RH103	1.21E 03	1.27E 03	1.33E 03	1.35E 03	1.36E 03	1.36E 03
RU104	1.37E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03
PD105	8.78E 02	8.82E 02	8.82E 02	8.82E 02	8.82E 02	8.82E 02
RU106	6.54E 02	6.18E 02	5.52E 02	4.93E 02	8.25E 01	6.72E-07
PD106	4.31E 02	4.67E 02	5.34E 02	5.93E 02	1.00E 03	1.08E 03
PD107	7.93E 02	7.93E 02	7.93E 02	7.93E 02	7.93E 02	7.93E 02
PD108	5.08E 02	5.08E 02	5.08E 02	5.08E 02	5.08E 02	5.08E 02
AG109	5.96E 02	5.97E 02	5.97E 02	5.97E 02	5.97E 02	5.97E 02
PD110	4.25E 00	4.25E 00	4.25E 00	4.25E 00	4.25E 00	4.25E 00
CD110	3.14E 01	3.15E 01	3.16E 01	3.17E 01	3.20E 01	3.21E 01
CD111	9.83E 01	1.00E 02	1.00E 02	1.00E 02	1.00E 02	1.00E 02
CD112	6.14E 01	6.16E 01	6.16E 01	6.16E 01	6.16E 01	6.16E 01
CD113	5.25E 00	5.25E 00	5.25E 00	5.25E 00	5.25E 00	5.25E 00
CD114	6.02E 00	6.02E 00	6.02E 00	6.02E 00	6.02E 00	6.02E 00
SN117	6.24E 00	6.24E 00	6.24E 00	6.24E 00	6.24E 00	6.24E 00
SB121	1.97E 01	1.98E 01	1.98E 01	1.98E 01	1.99E 01	2.04E 01
SB125	3.06E 01	3.07E 01	2.95E 01	2.83E 01	1.46E 01	1.42E-02
GD155	3.03E 01	3.35E 01	3.97E 01	4.55E 01	1.02E 02	1.35E 02
GD156	7.56E 01	7.83E 01	7.91E 01	7.92E 01	7.92E 01	7.92E 01
GD157	4.49E 01	4.50E 01	4.50E 01	4.50E 01	4.50E 01	4.50E 01
GD158	1.25E 02	1.25E 02	1.25E 02	1.25E 02	1.25E 02	1.25E 02
TB159	6.18E 01	6.19E 01	6.19E 01	6.19E 01	6.19E 01	6.19E 01
GD160	5.08E 00	5.08E 00	5.08E 00	5.08E 00	5.08E 00	5.08E 00
DY160	4.17E 00	4.65E 00	5.28E 00	5.63E 00	6.08E 00	6.08E 00
DY161	1.25E 01	1.27E 01	1.27E 01	1.27E 01	1.27E 01	1.27E 01
SUBTOT	5.44E 04	5.45E 04	5.45E 04	5.45E 04	5.45E 04	5.45E 04
TOTALS	5.46E 04	5.46E 04	5.46E 04	5.46E 04	5.46E 04	5.46E 04

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H	1.55E-01	1.54E-01	1.53E-01	1.51E-01	1.31E-01	2.85E-02
ZN	8.94E-04	1.95E-08	9.29E-18	0.0	0.0	0.0
GA	2.74E-04	8.49E-09	4.04E-18	0.0	0.0	0.0
GE	2.32E-01	2.34E-01	2.34E-01	2.34E-01	2.34E-01	2.34E-01
AS	1.96E-02	1.91E-02	1.91E-02	1.91E-02	1.91E-02	1.91E-02
SE	7.02E 00	7.02E 00	7.02E 00	7.02E 00	7.02E 00	7.02E 00
BR	2.42E 00	2.38E 00	2.38E 00	2.38E 00	2.38E 00	2.38E 00
KR	4.85E 02	4.85E 02	4.84E 02	4.84E 02	4.78E 02	4.50E 02
RB	2.67E 02	2.67E 02	2.68E 02	2.68E 02	2.74E 02	3.02E 02
SR	8.32E 02	8.14E 02	7.93E 02	7.82E 02	7.46E 02	5.33E 02
Y	4.03E 02	3.95E 02	3.84E 02	3.78E 02	3.71E 02	3.71E 02
ZR	4.33E 03	4.29E 03	4.25E 03	4.23E 03	4.23E 03	4.44E 03
NB	1.15E 02	1.07E 02	7.35E 01	4.39E 01	3.60E-03	1.22E-02
MO	5.39E 03	5.45E 03	5.56E 03	5.63E 03	5.71E 03	5.71E 03
TC	1.35E 03	1.36E 03	1.36E 03	1.36E 03	1.36E 03	1.36E 03
RU	4.96E 03	4.86E 03	4.74E 03	4.66E 03	4.24E 03	4.16E 03
RH	1.21E 03	1.27E 03	1.33E 03	1.35E 03	1.36E 03	1.36E 03
PD	2.61E 03	2.65E 03	2.72E 03	2.78E 03	3.19E 03	3.27E 03
AG	5.99E 02	5.98E 02	5.98E 02	5.97E 02	5.97E 02	5.97E 02
CD	2.05E 02	2.07E 02	2.07E 02	2.07E 02	2.07E 02	2.07E 02
IN	2.78E 00	2.81E 00	2.83E 00	2.84E 00	2.95E 00	3.56E 00
SN	1.19E 02	1.18E 02	1.18E 02	1.18E 02	1.18E 02	1.17E 02
SB	5.46E 01	5.23E 01	5.11E 01	4.99E 01	3.63E 01	2.23E 01
TE	8.83E 02	8.60E 02	8.51E 02	8.48E 02	8.56E 02	8.70E 02
I	7.20E 02	7.06E 02	7.15E 02	7.19E 02	7.25E 02	7.25E 02
XE	6.68E 03	6.69E 03	6.69E 03	6.69E 03	6.69E 03	6.69E 03
CS	6.03E 03	6.05E 03	6.04E 03	6.03E 03	5.90E 03	5.04E 03
BA	2.06E 03	2.02E 03	2.02E 03	2.03E 03	2.16E 03	3.03E 03
LA	1.79E 03	1.78E 03	1.78E 03	1.78E 03	1.78E 03	1.78E 03
CE	4.30E 03	4.23E 03	4.09E 03	4.01E 03	3.58E 03	3.53E 03
PR	1.67E 03	1.69E 03	1.74E 03	1.75E 03	1.76E 03	1.76E 03
ND	5.09E 03	5.17E 03	5.27E 03	5.34E 03	5.76E 03	5.81E 03
PM	5.79E 02	5.79E 02	5.56E 02	5.31E 02	2.68E 02	2.11E-01
SM	1.22E 03	1.24E 03	1.27E 03	1.29E 03	1.55E 03	1.77E 03
EU	2.35E 02	2.30E 02	2.24E 02	2.18E 02	1.66E 02	1.75E 02
GD	2.85E 02	2.91E 02	2.97E 02	3.03E 02	3.58E 02	3.98E 02
TB	6.39E 01	6.34E 01	6.27E 01	6.24E 01	6.19E 01	6.19E 01
DY	2.29E 01	2.39E 01	2.49E 01	2.55E 01	2.82E 01	2.86E 01
HO	3.66E-01	3.66E-01	3.66E-01	3.66E-01	3.66E-01	3.66E-01
ER	2.57E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01
TOTALS	5.46E 04	5.46E 04	5.46E 04	5.46E 04	5.46E 04	5.46E 04

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H 3	1.50E 03	1.49E 03	1.48E 03	1.47E 03	1.27E 03	2.77E 02
GA 72	8.37E 02	2.62E-02	1.25E-11	0.0	0.0	0.0
SE 79	4.16E-02	4.16E-02	4.16E-02	4.16E-02	4.16E-02	4.16E-02
KR 85	1.60E 04	1.59E 04	1.58E 04	1.56E 04	1.32E 04	2.32E 03
RB 86	5.18E 03	1.70E 03	1.83E 02	1.97E 01	1.12E-14	0.0
SR 89	1.46E 06	9.82E 05	4.41E 05	1.98E 05	6.62E-01	0.0
SR 90	6.67E 04	6.65E 04	6.63E 04	6.60E 04	6.19E 04	3.18E 04
Y 90	6.89E 04	6.66E 04	6.63E 04	6.60E 04	6.19E 04	3.18E 04
Y 91	2.02E 06	1.42E 06	7.00E 05	3.45E 05	4.98E 00	0.0
ZR 93	2.20E 00	2.20E 00	2.20E 00	2.20E 00	2.20E 00	2.20E 00
NB 93M	7.99E-02	8.87E-02	1.06E-01	1.24E-01	3.80E-01	1.74E 00
ZR 95	4.56E 06	3.31E 06	1.75E 06	9.21E 05	3.85E 01	0.0
NB 95M	9.11E 04	7.03E 04	3.72E 04	1.96E 04	8.18E-01	1.19E-12
NB 95	4.51E 06	4.21E 06	2.88E 06	1.72E 06	8.32E 01	0.0
MO 99	4.85E 06	2.83E 03	9.58E-04	3.25E-10	0.0	0.0
TC 99M	4.22E 06	2.70E 03	9.15E-04	3.10E-10	0.0	0.0
TC 99	2.29E 01	2.31E 01	2.31E 01	2.31E 01	2.31E 01	2.31E 01
RU103	4.66E 06	2.76E 06	9.64E 05	3.37E 05	2.17E-02	0.0
RH103M	4.66E 06	2.76E 06	9.65E 05	3.38E 05	2.17E-02	0.0
RH105	3.09E 06	2.83E 00	2.38E-12	0.0	0.0	0.0
RU106	2.20E 06	2.08E 06	1.85E 06	1.65E 06	2.77E 05	2.26E-03
RH106	2.20E 06	2.08E 06	1.85E 06	1.65E 06	2.77E 05	2.26E-03
AG110M	2.94E 03	2.71E 03	2.30E 03	1.95E 03	1.46E 02	2.69E-10
AG110	2.05E 05	3.53E 02	2.99E 02	2.54E 02	1.90E 01	3.50E-11
AG111	3.25E 05	2.03E 04	7.93E 01	3.10E-01	0.0	0.0
CD113M	2.16E 02	2.15E 02	2.13E 02	2.12E 02	1.86E 02	4.89E 01
IN114M	3.74E 00	2.47E 00	1.07E 00	4.68E-01	9.43E-07	0.0
IN114	6.39E 00	2.38E 00	1.04E 00	4.51E-01	9.10E-07	0.0
CD115M	6.39E 02	3.94E 02	1.50E 02	5.69E 01	1.36E-05	0.0
CD115	8.85E 03	7.91E-01	6.25E-09	4.93E-17	0.0	0.0
IN115M	8.85E 03	8.64E-01	6.82E-09	5.39E-17	0.0	0.0
SN117M	3.52E 00	7.97E-01	4.08E-02	2.09E-03	0.0	0.0
SN119M	3.29E 01	3.03E 01	2.57E 01	2.17E 01	1.58E 00	2.10E-12
SN121M	8.98E 01	8.97E 01	8.96E 01	8.94E 01	8.73E 01	6.83E 01
SB122	2.73E 03	1.62E 00	5.75E-07	2.04E-13	0.0	0.0
SN123M	1.32E 03	1.12E 03	8.01E 02	5.75E 02	3.03E 00	0.0
TE123M	6.00E 00	5.03E 00	3.52E 00	2.47E 00	9.09E-03	0.0
SB124	1.58E 02	1.12E 02	5.58E 01	2.79E 01	5.03E-04	0.0
SN125	9.82E 04	1.08E 04	1.29E 02	1.54E 00	0.0	0.0
SB125	3.24E 04	3.26E 04	3.13E 04	3.00E 04	1.54E 04	1.50E 01
TE125M	1.09E 04	1.14E 04	1.19E 04	1.20E 04	6.39E 03	6.23E 00
SN126	2.80E 00	2.80E 00	2.80E 00	2.80E 00	2.80E 00	2.80E 00
SB126M	3.76E 03	2.80E 00	2.80E 00	2.80E 00	2.80E 00	2.80E 00
SB126	7.02E 03	1.33E 03	5.06E 01	4.49E 00	2.77E 00	2.77E 00
SB127	5.43E 05	2.59E 03	5.69E-02	1.25E-06	0.0	0.0
TE127M	1.16E 05	9.98E 04	6.82E 04	4.66E 04	1.14E 02	0.0
TE127	5.38E 05	1.01E 05	6.74E 04	4.60E 04	1.12E 02	9.81E-06

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
TE129M	5.28E 05	2.88E 05	8.48E 04	2.49E 04	1.05E-04	0.0
TE129	1.27E 06	1.85E 05	5.43E 04	1.60E 04	6.73E-05	0.0
I129	8.33E-02	8.46E-02	8.57E-02	8.60E-02	8.62E-02	8.62E-02
TE131M	4.41E 05	2.66E-02	9.45E-17	0.0	0.0	0.0
I131	2.81E 06	2.18E 05	1.24E 03	7.10E 00	0.0	0.0
XE131M	2.92E 04	9.77E 03	3.77E 02	1.16E 01	0.0	0.0
TE132	3.93E 06	6.56E 03	1.87E-02	5.32E-08	0.0	0.0
I132	3.95E 06	6.76E 03	1.93E-02	5.49E-08	0.0	0.0
XE133M	1.28E 05	1.93E 01	1.96E-07	2.00E-15	0.0	0.0
XE133	5.13E 06	1.17E 05	4.38E 01	1.64E-02	0.0	0.0
CS134	4.99E 04	4.85E 04	4.59E 04	4.34E 04	1.81E 04	2.00E 00
CS135	1.85E 00	1.85E 00	1.85E 00	1.85E 00	1.85E 00	1.85E 00
CS136	2.34E 05	4.72E 04	1.93E 03	7.85E 01	0.0	0.0
CS137	1.71E 05	1.71E 05	1.70E 05	1.70E 05	1.60E 05	8.56E 04
BA137M	1.60E 05	1.60E 05	1.59E 05	1.59E 05	1.49E 05	8.01E 04
BA140	4.14E 06	8.16E 05	3.17E 04	1.23E 03	0.0	0.0
LA140	4.29E 06	9.39E 05	3.64E 04	1.41E 03	1.41E-15	0.0
CE141	4.39E 06	2.32E 06	6.43E 05	1.78E 05	2.89E-04	0.0
CE143	4.13E 06	1.13E 00	8.24E-14	0.0	0.0	0.0
PR143	4.15E 06	1.01E 06	4.85E 04	2.33E 03	3.81E-18	0.0
CE144	2.15E 06	2.00E 06	1.73E 06	1.49E 06	1.48E 05	5.22E-06
PR144	2.25E 06	2.00E 06	1.73E 06	1.49E 06	1.48E 05	5.22E-06
ND147	1.85E 06	2.85E 05	6.72E 03	1.59E 02	0.0	0.0
PM147	5.28E 05	5.35E 05	5.15E 05	4.93E 05	2.49E 05	1.96E 02
PM148M	1.12E 05	6.86E 04	2.55E 04	9.46E 03	1.57E-03	0.0
PM148	1.34E 05	8.16E 03	2.05E 03	7.60E 02	1.25E-04	0.0
PM149	1.11E 06	9.55E 01	6.55E-07	4.50E-15	0.0	0.0
SM151	7.09E 03	7.11E 03	7.10E 03	7.09E 03	6.94E 03	5.60E 03
EU152	1.64E 01	1.63E 01	1.62E 01	1.60E 01	1.38E 01	2.90E 00
SM153	3.06E 05	7.55E 00	4.52E-09	0.0	0.0	0.0
GD153	9.18E-01	8.43E-01	7.09E-01	5.97E-01	3.98E-02	2.14E-14
EU154	1.61E 03	1.60E 03	1.59E 03	1.58E 03	1.41E 03	4.38E 02
EU155	1.33E 05	1.29E 05	1.21E 05	1.14E 05	4.23E 04	1.37E 00
EU156	1.95E 05	4.98E 04	3.11E 03	1.94E 02	2.01E-17	0.0
TB160	2.16E 04	1.62E 04	9.10E 03	5.11E 03	5.73E-01	0.0
TB161	3.03E 04	1.49E 03	3.59E 00	8.65E-03	0.0	0.0
GD162	7.73E 03	7.30E 03	6.51E 03	5.81E 03	9.65E 02	7.19E-06
TB162M	7.73E 03	7.30E 03	6.51E 03	5.81E 03	9.65E 02	7.19E-06
DY166	4.54E 02	9.95E-01	4.84E-06	2.35E-11	0.0	0.0
HO166	6.10E 02	1.49E 00	7.22E-06	3.51E-11	0.0	0.0
SUBTOT	8.94E 07	3.16E 07	1.72E 07	1.17E 07	1.64E 06	2.38E 05
TOTALS	4.04E 08	3.16E 07	1.72E 07	1.17E 07	1.64E 06	2.38E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H 3	5.34E-02	5.31E-02	5.26E-02	5.21E-02	4.51E-02	9.84E-03
KR 85	2.57E 01	2.56E 01	2.53E 01	2.50E 01	2.12E 01	3.72E 00
RB 86	2.44E 01	8.00E 00	8.61E-01	9.27E-02	5.25E-17	0.0
SR 89	5.27E 03	3.53E 03	1.59E 03	7.14E 02	2.38E-03	0.0
SR 90	8.70E 01	8.68E 01	8.64E 01	8.61E 01	8.08E 01	4.15E 01
Y 90	3.94E 02	3.81E 02	3.79E 02	3.78E 02	3.54E 02	1.82E 02
Y 91	7.68E 03	5.40E 03	2.66E 03	1.31E 03	1.89E-02	0.0
ZR 95	2.39E 04	1.73E 04	9.14E 03	4.82E 03	2.02E-01	0.0
NB 95M	1.27E 02	9.79E 01	5.18E 01	2.72E 01	1.14E-03	1.66E-15
NB 95	2.16E 04	2.02E 04	1.38E 04	8.26E 03	3.99E-01	0.0
MO 99	2.18E 04	1.27E 01	4.30E-06	1.46E-12	0.0	0.0
TC 99M	3.58E 03	2.29E 00	7.76E-07	2.63E-13	0.0	0.0
RU103	1.52E 04	9.00E 03	3.15E 03	1.10E 03	7.09E-05	0.0
RH103M	1.10E 03	6.54E 02	2.29E 02	8.00E 01	5.15E-06	0.0
RU106	1.30E 02	1.23E 02	1.10E 02	9.81E 01	1.64E 01	1.34E-07
RH106	2.12E 04	2.00E 04	1.79E 04	1.60E 04	2.68E 03	2.18E-05
AG110M	4.75E 01	4.37E 01	3.71E 01	3.15E 01	2.36E 00	4.34E-12
AG110	1.48E 03	2.56E 00	2.17E 00	1.84E 00	1.38E-01	2.54E-13
AG111	7.78E 02	4.86E 01	1.90E-01	7.42E-04	0.0	0.0
CD113M	2.86E-01	2.84E-01	2.82E-01	2.80E-01	2.46E-01	6.47E-02
CD115M	2.60E 00	1.60E 00	6.10E-01	2.32E-01	5.53E-08	0.0
SN121M	9.42E-02	9.41E-02	9.40E-02	9.38E-02	9.16E-02	7.16E-02
SN123M	4.50E 00	3.81E 00	2.73E 00	1.96E 00	1.03E-02	0.0
SB124	2.13E 00	1.51E 00	7.54E-01	3.77E-01	6.80E-06	0.0
SN125	5.95E 02	6.51E 01	7.80E-01	9.35E-03	0.0	0.0
SB125	1.09E 02	1.09E 02	1.05E 02	1.01E 02	5.18E 01	5.05E-02
TE125M	9.36E 00	9.83E 00	1.03E 01	1.03E 01	5.50E 00	5.36E-03
SB126	7.70E 01	1.46E 01	5.54E-01	4.92E-02	3.04E-02	3.04E-02
SB127	2.98E 03	1.42E 01	3.13E-04	6.87E-09	0.0	0.0
TE127M	6.41E 01	5.50E 01	3.76E 01	2.57E 01	6.26E-02	0.0
TE127	8.77E 02	1.65E 02	1.10E 02	7.50E 01	1.33E-01	1.60E-08
TE129M	1.05E 03	5.70E 02	1.68E 02	4.94E 01	2.08E-07	0.0
TE129	4.61E 03	6.70E 02	1.97E 02	5.80E 01	2.44E-07	0.0
I131	1.15E 04	8.98E 02	5.12E 00	2.92E-02	0.0	0.0
XE131M	2.84E 01	9.50E 00	3.66E-01	1.13E-02	0.0	0.0
TE132	5.36E 03	8.94E 00	2.55E-05	7.26E-11	0.0	0.0
I132	6.32E 04	1.08E 02	3.08E-04	8.78E-10	0.0	0.0
XE133M	1.77E 02	2.66E-02	2.71E-10	2.76E-18	0.0	0.0
XE133	5.54E 03	1.26E 02	4.73E-02	1.77E-05	0.0	0.0
CS134	5.23E 02	5.09E 02	4.82E 02	4.56E 02	1.90E 02	2.10E-02
CS136	3.62E 03	7.30E 02	2.98E 01	1.22E 00	0.0	0.0
CS137	2.77E 02	2.77E 02	2.76E 02	2.75E 02	2.59E 02	1.39E 02
BA137M	6.28E 02	6.27E 02	6.25E 02	6.23E 02	5.86E 02	3.14E 02
BA140	1.40E 04	2.75E 03	1.07E 02	4.14E 00	0.0	0.0
LA140	7.13E 04	1.56E 04	6.05E 02	2.35E 01	2.34E-17	0.0
CE141	8.63E 03	4.57E 03	1.27E 03	3.51E 02	5.59E-07	0.0
PR143	8.99E 03	2.19E 03	1.05E 02	5.06E 00	8.27E-21	0.0

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
CE144	1.89E 03	1.76E 03	1.52E 03	1.31E 03	1.30E 02	4.58E-09
PR144	1.67E 04	1.49E 04	1.29E 04	1.11E 04	1.10E 03	3.89E-08
ND147	5.18E 03	7.97E 02	1.88E 01	4.44E-01	0.0	0.0
PM147	2.72E 02	2.76E 02	2.66E 02	2.54E 02	1.28E 02	1.01E-01
PM148M	1.43E 03	8.71E 02	3.23E 02	1.20E 02	1.99E-05	0.0
PM148	1.10E 03	6.68E 01	1.67E 01	6.22E 00	1.03E-06	0.0
PM149	2.78E 03	2.38E-01	1.64E-09	1.12E-17	0.0	0.0
SM151	1.24E 01	1.24E 01	1.24E 01	1.24E 01	1.21E 01	9.76E 00
EU152	2.00E-01	1.99E-01	1.97E-01	1.96E-01	1.63E-01	3.54E-02
EU154	1.51E 01	1.50E 01	1.49E 01	1.48E 01	1.32E 01	4.11E 00
EU155	1.12E 02	1.09E 02	1.02E 02	9.59E 01	3.55E 01	1.15E-03
EU156	2.06E 03	5.25E 02	3.28E 01	2.05E 00	2.12E-19	0.0
TB160	1.82E 02	1.36E 02	7.66E 01	4.30E 01	4.83E-03	0.0
TB161	4.94E 01	2.43E 00	5.85E-03	1.41E-05	0.0	0.0
GD162	2.63E 01	2.48E 01	2.22E 01	1.98E 01	3.28E 00	2.45E-08
TB162M	5.18E 01	4.89E 01	4.36E 01	3.89E 01	6.47E 00	4.81E-08
SUBTOT	3.60E 05	1.27E 05	6.86E 04	4.80E 04	5.68E 03	6.94E 02
TOTALS	4.14E 06	1.27E 05	6.86E 04	4.80E 04	5.63E 03	6.94E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30 D	90. D	150. D	1096. D	10958. D
KR 85	2.06E-01	2.05E-01	2.02E-01	2.00E-01	1.70E-01	2.98E-02
RB 86	2.83E 00	9.28E-01	9.99E-02	1.08E-02	6.09E-18	0.0
Y 91	4.61E 01	3.24E 01	1.60E 01	7.88E 00	1.14E-04	0.0
ZR 95	1.96E 04	1.42E 04	7.51E 03	3.96E 03	1.66E-01	0.0
NB 95M	1.27E 02	9.79E 01	5.18E 01	2.72E 01	1.14E-03	1.66E-15
NB 95	2.04E 04	1.91E 04	1.31E 04	7.81E 03	3.77E-01	0.0
MO 99	6.32E 03	3.68E 00	1.25E-06	4.23E-13	0.0	0.0
RU103	1.34E 04	7.92E 03	2.77E 03	9.69E 02	6.24E-05	0.0
RH106	2.55E 03	2.41E 03	2.15E 03	1.92E 03	3.21E 02	2.62E-06
AG110M	4.56E 01	4.20E 01	3.56E 01	3.02E 01	2.26E 00	4.16E-12
AG110	4.01E 01	6.91E-02	5.86E-02	4.97E-02	3.72E-03	6.85E-15
CD115M	3.82E-01	2.36E-01	8.97E-02	3.41E-02	8.12E-09	0.0
SB124	1.75E 00	1.24E 00	6.18E-01	3.09E-01	5.58E-06	0.0
SN125	5.41E 01	5.92E 00	7.10E-02	8.50E-04	0.0	0.0
SB125	5.98E 01	6.01E 01	5.78E 01	5.54E 01	2.85E 01	2.78E-02
SB126	4.47E 01	8.48E 00	3.22E-01	2.85E-02	1.76E-02	1.76E-02
SB127	1.51E 03	7.22E 00	1.59E-04	3.49E-09	0.0	0.0
TE127	1.32E 01	2.47E 00	1.65E 00	1.13E 00	2.75E-03	2.40E-10
TE129M	1.25E 02	6.84E 01	2.01E 01	5.93E 00	2.49E-08	0.0
TE129	6.91E 02	1.00E 02	2.96E 01	8.70E 00	3.66E-08	0.0
I131	6.63E 03	5.15E 02	2.94E 00	1.68E-02	0.0	0.0
TE132	2.04E 03	3.40E 00	9.68E-06	2.76E-11	0.0	0.0
I132	4.93E 04	8.44E 01	2.40E-04	6.85E-10	0.0	0.0
CS134	4.66E 02	4.53E 02	4.29E 02	4.05E 02	1.69E 02	1.87E-02
CS136	3.40E 03	6.87E 02	2.80E 01	1.14E 00	0.0	0.0
BA137M	6.28E 02	6.27E 02	6.25E 02	6.23E 02	5.85E 02	3.14E 02
BA140	6.56E 03	1.29E 03	5.02E 01	1.95E 00	0.0	0.0
LA140	5.49E 04	1.20E 04	4.66E 02	1.81E 01	1.81E-17	0.0
CE144	5.10E 02	4.74E 02	4.09E 02	3.54E 02	3.51E 01	1.24E-09
PR144	4.02E 02	3.57E 02	3.09E 02	2.67E 02	2.65E 01	9.33E-10
ND147	2.23E 02	3.43E 01	8.08E-01	1.91E-02	0.0	0.0
PM148M	1.30E 03	7.92E 02	2.94E 02	1.09E 02	1.81E-05	0.0
PM148	5.04E 02	3.07E 01	7.70E 00	2.86E 00	4.74E-07	0.0
EU154	1.27E 01	1.26E 01	1.25E 01	1.24E 01	1.11E 01	3.45E 00
EU156	1.56E 03	3.99E 02	2.49E 01	1.56E 00	1.61E-19	0.0
TB160	1.49E 02	1.12E 02	6.28E 01	3.53E 01	3.96E-03	0.0
SUBTOT	1.94E 05	6.20E 04	2.84E 04	1.66E 04	1.18E 03	3.18E 02
TOTALS	1.44E 06	6.20E 04	2.84E 04	1.66E 04	1.18E 03	3.18E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H	5.34E-02	5.31E-02	5.26E-02	5.21E-02	4.51E-02	9.84E-03
ZN	1.22E 00	2.66E-05	1.27E-14	0.0	0.0	0.0
GA	2.36E 01	5.38E-04	2.56E-13	0.0	0.0	0.0
GE	8.42E 00	1.21E-19	0.0	0.0	0.0	0.0
AS	1.48E 02	1.51E-06	9.49E-18	0.0	0.0	0.0
SE	8.93E 03	1.58E-05	1.58E-05	1.58E-05	1.58E-05	1.58E-05
BR	8.47E 04	4.83E-06	2.62E-18	0.0	0.0	0.0
KR	1.26E 05	2.56E 01	2.53E 01	2.50E 01	2.12E 01	3.72E 00
RB	3.43E 05	8.00E 00	8.61E-01	9.27E-02	1.07E-08	1.07E-08
SR	2.13E 05	3.62E 03	1.67E 03	8.00E 02	8.08E 01	4.15E 01
Y	3.09E 05	5.79E 03	3.04E 03	1.69E 03	3.54E 02	1.82E 02
ZR	5.44E 04	1.73E 04	9.14E 03	4.82E 03	2.02E-01	2.61E-04
NB	2.75E 05	2.03E 04	1.39E 04	8.29E 03	4.00E-01	3.09E-04
MO	1.75E 05	1.27E 01	4.30E-06	1.46E-12	0.0	0.0
TC	2.64E 05	2.30E 00	1.56E-02	1.56E-02	1.56E-02	1.56E-02
RU	6.44E 04	9.12E 03	3.26E 03	1.20E 03	1.64E 01	1.34E-07
RH	7.89E 04	2.07E 04	1.81E 04	1.61E 04	2.68E 03	2.18E-05
PD	7.68E 03	4.61E-09	0.0	0.0	0.0	0.0
AG	6.50E 03	9.49E 01	3.95E 01	3.33E 01	2.49E 00	4.59E-12
CD	4.37E 02	1.89E 00	8.92E-01	5.12E-01	2.46E-01	6.47E-02
IN	8.79E 02	1.64E-02	6.41E-03	2.79E-03	5.62E-09	0.0
SN	1.92E 04	6.90E 01	3.62E 00	2.07E 00	1.04E-01	7.31E-02
SB	2.59E 05	1.40E 02	1.06E 02	1.01E 02	5.18E 01	8.25E-02
TE	1.96E 05	1.48E 03	5.23E 02	2.18E 02	5.74E 00	5.36E-03
I	4.03E 05	1.01E 03	5.12E 00	2.92E-02	3.73E-05	3.73E-05
XE	2.46E 05	1.36E 02	4.14E-01	1.13E-02	0.0	0.0
CS	4.02E 05	1.52E 03	7.87E 02	7.31E 02	4.48E 02	1.39E 02
BA	1.39E 05	3.38E 03	7.32E 02	6.27E 02	5.86E 02	3.14E 02
LA	2.49E 05	1.56E 04	6.05E 02	2.35E 01	2.34E-17	0.0
CE	7.61E 04	6.32E 03	2.78E 03	1.66E 03	1.30E 02	4.58E-09
PR	9.72E 04	1.71E 04	1.30E 04	1.11E 04	1.10E 03	3.89E-08
ND	1.57E 04	7.97E 02	1.88E 01	4.44E-01	0.0	0.0
PM	1.34E 04	1.21E 03	6.06E 02	3.81E 02	1.28E 02	1.01E-01
SM	2.31E 03	1.24E 01	1.24E 01	1.24E 01	1.21E 01	9.76E 00
EU	6.62E 03	6.49E 02	1.50E 02	1.13E 02	4.90E 01	4.14E 00
GD	5.70E 02	2.48E 01	2.22E 01	1.98E 01	3.28E 00	2.45E-08
TB	2.97E 02	1.88E 02	1.20E 02	8.19E 01	6.47E 00	4.81E-08
DY	3.34E 00	8.43E-04	4.10E-09	1.99E-14	0.0	0.0
HO	2.67E 00	6.51E-03	3.16E-08	1.54E-13	0.0	0.0
TOTALS	4.14E 06	1.27E 05	6.86E 04	4.80E 04	5.68E 03	6.94E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
SE 82	4.30E 00	4.30E 00	4.30E 00	4.30E 00	4.30E 00	4.30E 00
KR 83	6.54E 01	6.54E 01	6.54E 01	6.54E 01	6.54E 01	6.54E 01
KR 84	1.14E 02	1.14E 02	1.14E 02	1.14E 02	1.14E 02	1.14E 02
KR 85	2.62E 01	2.61E 01	2.58E 01	2.56E 01	2.16E 01	3.80E 00
RB 85	4.39E 01	4.41E 01	4.43E 01	4.46E 01	4.85E 01	6.64E 01
KR 86	1.08E 02	1.08E 02	1.08E 02	1.08E 02	1.08E 02	1.08E 02
RB 87	1.34E 02	1.34E 02	1.34E 02	1.34E 02	1.34E 02	1.34E 02
SR 88	2.04E 02	2.04E 02	2.04E 02	2.04E 02	2.04E 02	2.04E 02
SR 89	3.37E 01	2.26E 01	1.01E 01	4.56E 00	1.52E-05	0.0
Y 89	2.12E 02	2.23E 02	2.35E 02	2.41E 02	2.46E 02	2.46E 02
SR 90	3.08E 02	3.07E 02	3.06E 02	3.05E 02	2.86E 02	1.47E 02
ZR 90	6.03E 00	6.66E 00	7.90E 00	9.14E 00	2.80E 01	1.67E 02
Y 91	5.36E 01	3.77E 01	1.86E 01	9.17E 00	1.32E-04	0.0
ZR 91	2.93E 02	3.09E 02	3.28E 02	3.38E 02	3.47E 02	3.47E 02
ZR 92	4.49E 02	4.50E 02	4.50E 02	4.50E 02	4.50E 02	4.50E 02
ZR 93	5.55E 02	5.56E 02	5.56E 02	5.56E 02	5.56E 02	5.56E 02
ZR 94	6.30E 02	6.30E 02	6.30E 02	6.30E 02	6.30E 02	6.30E 02
ZR 95	1.37E 02	9.92E 01	5.23E 01	2.76E 01	1.15E-03	0.0
NB 95	7.25E 01	6.78E 01	4.65E 01	2.78E 01	1.34E-03	0.0
MO 95	5.75E 02	6.18E 02	6.86E 02	7.29E 02	7.84E 02	7.84E 02
ZR 96	7.34E 02	7.34E 02	7.34E 02	7.34E 02	7.34E 02	7.34E 02
MO 96	1.30E 01	1.30E 01	1.30E 01	1.30E 01	1.30E 01	1.30E 01
MO 97	9.82E 02	9.84E 02	9.84E 02	9.84E 02	9.84E 02	9.84E 02
MO 98	8.56E 02	8.56E 02	8.56E 02	8.56E 02	8.56E 02	8.56E 02
TC 99	8.63E 02	8.71E 02	8.71E 02	8.70E 02	8.70E 02	8.70E 02
MO100	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03
RU100	2.03E 01	2.03E 01	2.03E 01	2.03E 01	2.03E 01	2.03E 01
RU101	8.34E 02	8.34E 02	8.34E 02	8.34E 02	8.34E 02	8.34E 02
RU102	9.36E 02	9.36E 02	9.36E 02	9.36E 02	9.36E 02	9.36E 02
RU103	9.29E 01	5.49E 01	1.92E 01	6.72E 00	4.33E-07	0.0
RH103	7.81E 02	8.19E 02	8.55E 02	8.68E 02	8.74E 02	8.74E 02
RU104	8.70E 02	8.70E 02	8.70E 02	8.70E 02	8.70E 02	8.70E 02
PD105	5.58E 02	5.61E 02	5.61E 02	5.61E 02	5.61E 02	5.61E 02
RU106	4.07E 02	3.84E 02	3.43E 02	3.06E 02	5.13E 01	4.18E-07
PD106	2.71E 02	2.94E 02	3.35E 02	3.72E 02	6.27E 02	6.78E 02
PD107	4.96E 02	4.96E 02	4.96E 02	4.96E 02	4.96E 02	4.96E 02
PD108	3.11E 02	3.11E 02	3.11E 02	3.11E 02	3.11E 02	3.11E 02
AG109	3.62E 02	3.63E 02	3.63E 02	3.63E 02	3.63E 02	3.63E 02
CD110	1.84E 01	1.84E 01	1.85E 01	1.85E 01	1.87E 01	1.88E 01
CD111	6.00E 01	6.13E 01	6.13E 01	6.13E 01	6.13E 01	6.13E 01
CD112	3.73E 01	3.74E 01	3.74E 01	3.74E 01	3.74E 01	3.74E 01
CD114	4.01E 00	4.01E 00	4.01E 00	4.01E 00	4.01E 00	4.01E 00
SN117	4.16E 00	4.16E 00	4.16E 00	4.16E 00	4.16E 00	4.16E 00
SB121	1.23E 01	1.24E 01	1.24E 01	1.24E 01	1.24E 01	1.27E 01
SB125	1.90E 01	1.90E 01	1.83E 01	1.75E 01	9.02E 00	8.80E-03
SN126	5.99E 01	5.99E 01	5.99E 01	5.99E 01	5.99E 01	5.99E 01
TE127M	7.54E 00	6.47E 00	4.42E 00	3.02E 00	7.36E-03	0.0

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
I127	1.11E 02	1.13E 02	1.15E 02	1.17E 02	1.20E 02	1.20E 02
TE128	1.61E 02	1.61E 02	1.61E 02	1.61E 02	1.61E 02	1.61E 02
TE129M	1.12E 01	6.08E 00	1.79E 00	5.27E-01	2.22E-09	0.0
I129	3.21E 02	3.27E 02	3.31E 02	3.32E 02	3.33E 02	3.33E 02
TE130	3.65E 02	3.65E 02	3.65E 02	3.65E 02	3.65E 02	3.65E 02
XE130	9.74E 00	9.77E 00	9.77E 00	9.77E 00	9.77E 00	9.77E 00
XE131	6.65E 02	6.79E 02	6.80E 02	6.80E 02	6.80E 02	6.80E 02
XE132	9.68E 02	9.77E 02	9.77E 02	9.77E 02	9.77E 02	9.77E 02
CS133	1.22E 03	1.24E 03	1.24E 03	1.24E 03	1.24E 03	1.24E 03
XE134	1.37E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03	1.37E 03
CS134	2.29E 01	2.22E 01	2.10E 01	1.99E 01	8.29E 00	9.19E-04
BA134	3.99E 00	4.62E 00	5.82E 00	6.95E 00	1.86E 01	2.69E 01
CS135	1.33E 03	1.33E 03	1.33E 03	1.33E 03	1.33E 03	1.33E 03
XE136	1.22E 03	1.22E 03	1.22E 03	1.22E 03	1.22E 03	1.22E 03
BA136	3.64E 01	3.79E 01	3.83E 01	3.83E 01	3.83E 01	3.83E 01
CS137	1.26E 03	1.25E 03	1.25E 03	1.24E 03	1.17E 03	6.28E 02
BA137	2.21E 01	2.45E 01	2.93E 01	3.40E 01	1.06E 02	6.50E 02
BA138	1.22E 03	1.22E 03	1.22E 03	1.22E 03	1.22E 03	1.22E 03
LA139	1.14E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03
BA140	3.64E 01	7.17E 00	2.78E-01	1.08E-02	0.0	0.0
CE140	1.21E 03	1.25E 03	1.26E 03	1.26E 03	1.26E 03	1.26E 03
CE141	9.78E 01	5.18E 01	1.43E 01	3.97E 00	6.45E-09	0.0
PR141	1.03E 03	1.08E 03	1.12E 03	1.13E 03	1.13E 03	1.13E 03
CE142	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03	1.01E 03
ND142	4.76E 00	4.78E 00	4.78E 00	4.78E 00	4.78E 00	4.78E 00
PR143	3.96E 01	9.66E 00	4.64E-01	2.23E-02	3.64E-23	0.0
ND143	9.97E 02	1.03E 03	1.04E 03	1.04E 03	1.04E 03	1.04E 03
CE144	4.30E 02	4.00E 02	3.45E 02	2.98E 02	2.96E 01	1.04E-09
ND144	4.15E 02	4.46E 02	5.00E 02	5.47E 02	8.16E 02	8.46E 02
ND145	6.64E 02	6.65E 02	6.65E 02	6.65E 02	6.65E 02	6.65E 02
ND146	6.17E 02	6.17E 02	6.17E 02	6.17E 02	6.17E 02	6.17E 02
PM147	3.76E 02	3.80E 02	3.66E 02	3.51E 02	1.77E 02	1.40E-01
SM147	7.89E 01	8.72E 01	1.03E 02	1.19E 02	2.93E 02	4.69E 02
ND148	3.83E 02	3.83E 02	3.83E 02	3.83E 02	3.83E 02	3.83E 02
SM148	3.27E 01	3.44E 01	3.57E 01	3.62E 01	3.65E 01	3.65E 01
SM149	2.81E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02
ND150	2.26E 02	2.26E 02	2.26E 02	2.26E 02	2.26E 02	2.26E 02
SM150	2.07E 01	2.07E 01	2.07E 01	2.07E 01	2.07E 01	2.07E 01
SM151	1.72E 02	1.72E 02	1.72E 02	1.72E 02	1.68E 02	1.36E 02
SM152	1.53E 02	1.53E 02	1.53E 02	1.53E 02	1.53E 02	1.53E 02
EU153	7.43E 01	7.48E 01	7.48E 01	7.48E 01	7.48E 01	7.48E 01
SM154	6.04E 01	6.04E 01	6.04E 01	6.04E 01	6.04E 01	6.04E 01
EU154	6.75E 00	6.72E 00	6.68E 00	6.63E 00	5.92E 00	1.84E 00
EU155	6.43E 01	6.23E 01	5.85E 01	5.49E 01	2.04E 01	6.58E-04
GD155	1.88E 01	2.08E 01	2.46E 01	2.82E 01	6.28E 01	8.31E 01
GD156	4.65E 01	4.81E 01	4.86E 01	4.87E 01	4.87E 01	4.87E 01
GD157	2.75E 01	2.76E 01	2.76E 01	2.76E 01	2.76E 01	2.76E 01
GD158	7.54E 01	7.54E 01	7.54E 01	7.54E 01	7.54E 01	7.54E 01
TB159	3.74E 01	3.75E 01	3.75E 01	3.75E 01	3.75E 01	3.75E 01
DY161	7.53E 00	7.63E 00	7.69E 00	7.69E 00	7.69E 00	7.69E 00
SUBTOT	3.48E 04	3.49E 04	3.49E 04	3.49E 04	3.49E 04	3.48E 04
TOTALS	3.49E 04	3.49E 04	3.49E 04	3.49E 04	3.49E 04	3.49E 04

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H	9.66E-02	9.61E-02	9.53E-02	9.44E-02	8.16E-02	1.78E-02
ZN	5.51E-04	1.20E-08	5.73E-18	0.0	0.0	0.0
GA	1.69E-04	5.24E-09	2.50E-18	0.0	0.0	0.0
GE	1.59E-01	1.61E-01	1.61E-01	1.61E-01	1.61E-01	1.61E-01
AS	1.90E-02	1.86E-02	1.86E-02	1.86E-02	1.86E-02	1.86E-02
SE	6.57E 00	6.57E 00	6.57E 00	6.57E 00	6.57E 00	6.57E 00
BR	2.26E 00	2.23E 00	2.23E 00	2.23E 00	2.23E 00	2.23E 00
KR	3.13E 02	3.13E 02	3.13E 02	3.12E 02	3.08E 02	2.91E 02
RB	1.78E 02	1.78E 02	1.78E 02	1.79E 02	1.82E 02	2.00E 02
SR	5.46E 02	5.34E 02	5.21E 02	5.14E 02	4.90E 02	3.51E 02
Y	2.66E 02	2.61E 02	2.54E 02	2.50E 02	2.46E 02	2.46E 02
ZR	2.81E 03	2.78E 03	2.76E 03	2.74E 03	2.74E 03	2.88E 03
NB	7.29E 01	6.79E 01	4.65E 01	2.78E 01	2.31E-03	7.90E-03
MO	3.44E 03	3.48E 03	3.55E 03	3.59E 03	3.64E 03	3.64E 03
TC	8.64E 02	8.71E 02	8.71E 02	8.70E 02	8.70E 02	8.70E 02
RU	3.16E 03	3.10E 03	3.02E 03	2.97E 03	2.71E 03	2.66E 03
RH	7.84E 02	8.19E 02	8.55E 02	8.68E 02	8.74E 02	8.74E 02
PD	1.64E 03	1.67E 03	1.71E 03	1.74E 03	2.00E 03	2.05E 03
AG	3.64E 02	3.64E 02	3.63E 02	3.63E 02	3.63E 02	3.63E 02
CD	1.25E 02	1.26E 02	1.26E 02	1.26E 02	1.27E 02	1.26E 02
IN	2.01E 00	2.03E 00	2.04E 00	2.05E 00	2.12E 00	2.47E 00
SN	7.48E 01	7.42E 01	7.41E 01	7.41E 01	7.40E 01	7.37E 01
SB	3.44E 01	3.30E 01	3.22E 01	3.15E 01	2.30E 01	1.43E 01
TE	5.60E 02	5.45E 02	5.40E 02	5.38E 02	5.43E 02	5.52E 02
I	4.50E 02	4.41E 02	4.46E 02	4.49E 02	4.52E 02	4.52E 02
XE	4.26E 03	4.26E 03	4.26E 03	4.26E 03	4.26E 03	4.26E 03
CS	3.83E 03	3.85E 03	3.84E 03	3.84E 03	3.75E 03	3.20E 03
BA	1.32E 03	1.29E 03	1.29E 03	1.30E 03	1.38E 03	1.93E 03
LA	1.15E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03	1.14E 03
CE	2.75E 03	2.70E 03	2.62E 03	2.56E 03	2.29E 03	2.26E 03
PR	1.07E 03	1.09E 03	1.12E 03	1.13E 03	1.13E 03	1.13E 03
ND	3.32E 03	3.38E 03	3.44E 03	3.49E 03	3.75E 03	3.78E 03
PM	3.82E 02	3.82E 02	3.67E 02	3.51E 02	1.77E 02	1.40E-01
SM	8.00E 02	8.12E 02	8.29E 02	8.45E 02	1.02E 03	1.16E 03
EU	1.49E 02	1.45E 02	1.41E 02	1.38E 02	1.06E 02	1.14E 02
GD	1.74E 02	1.77E 02	1.81E 02	1.85E 02	2.19E 02	2.43E 02
TB	3.86E 01	3.83E 01	3.79E 01	3.77E 01	3.75E 01	3.75E 01
DY	1.38E 01	1.43E 01	1.49E 01	1.53E 01	1.69E 01	1.71E 01
HO	2.22E-01	2.21E-01	2.21E-01	2.21E-01	2.21E-01	2.21E-01
ER	1.54E-01	1.56E-01	1.56E-01	1.56E-01	1.55E-01	1.56E-01
TOTALS	3.49E 04	3.49E 04	3.49E 04	3.49E 04	3.49E 04	3.49E 04

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H 3	9.36E 02	9.32E 02	9.23E 02	9.15E 02	7.91E 02	1.73E 02
SE 79	3.92E-02	3.92E-02	3.92E-02	3.92E-02	3.92E-02	3.92E-02
KR 85	1.03E 04	1.02E 04	1.01E 04	1.00E 04	8.46E 03	1.49E 03
RB 86	3.12E 03	1.02E 03	1.10E 02	1.19E 01	6.72E-15	0.0
SR 89	9.50E 05	6.37E 05	2.86E 05	1.29E 05	4.30E-01	0.0
SR 90	4.35E 04	4.34E 04	4.33E 04	4.31E 04	4.04E 04	2.08E 04
Y 90	4.48E 04	4.35E 04	4.33E 04	4.31E 04	4.04E 04	2.08E 04
Y 91	1.31E 06	9.21E 05	4.54E 05	2.24E 05	3.23E 00	0.0
ZR 93	1.42E 00	1.43E 00	1.43E 00	1.43E 00	1.43E 00	1.43E 00
NB 93M	5.39E-02	5.96E-02	7.10E-02	8.23E-02	2.48E-01	1.13E 00
ZR 95	2.89E 06	2.10E 06	1.11E 06	5.84E 05	2.44E 01	0.0
NB 95M	5.77E 04	4.45E 04	2.36E 04	1.24E 04	5.18E-01	7.53E-13
NB 95	2.85E 06	2.66E 06	1.82E 06	1.09E 06	5.27E 01	0.0
MO 99	3.10E 06	1.81E 03	6.12E-04	2.07E-10	0.0	0.0
TC 99M	2.70E 06	1.73E 03	5.85E-04	1.98E-10	0.0	0.0
TC 99	1.47E 01	1.48E 01	1.48E 01	1.48E 01	1.48E 01	1.48E 01
RU103	2.97E 06	1.76E 06	6.15E 05	2.15E 05	1.39E-02	0.0
RH103M	2.97E 06	1.76E 06	6.16E 05	2.16E 05	1.39E-02	0.0
RH105	1.96E 06	1.80E 00	1.51E-12	0.0	0.0	0.0
RU106	1.37E 06	1.29E 06	1.15E 06	1.03E 06	1.72E 05	1.40E-03
RH106	1.37E 06	1.29E 06	1.15E 06	1.03E 06	1.72E 05	1.40E-03
AG110M	1.72E 03	1.59E 03	1.35E 03	1.14E 03	8.55E 01	1.57E-10
AG110	1.20E 05	2.06E 02	1.75E 02	1.48E 02	1.11E 01	2.05E-11
AG111	2.01E 05	1.26E 04	4.91E 01	1.92E-01	0.0	0.0
CD113M	1.27E 02	1.26E 02	1.25E 02	1.24E 02	1.09E 02	2.86E 01
IN114M	2.17E 00	1.43E 00	6.22E-01	2.71E-01	5.46E-07	0.0
IN114	3.70E 00	1.38E 00	6.00E-01	2.61E-01	5.26E-07	0.0
CD115M	4.37E 02	2.69E 02	1.02E 02	3.89E 01	9.28E-06	0.0
CD115	6.10E 03	5.45E-01	4.30E-09	3.40E-17	0.0	0.0
IN115M	6.10E 03	5.95E-01	4.70E-09	3.71E-17	0.0	0.0
SN117M	2.10E 00	4.76E-01	2.44E-02	1.25E-03	0.0	0.0
SN119M	2.28E 01	2.10E 01	1.78E 01	1.50E 01	1.09E 00	1.45E-12
SN121M	5.41E 01	5.40E 01	5.40E 01	5.39E 01	5.26E 01	4.11E 01
SB122	1.61E 03	9.57E-01	3.39E-07	1.20E-13	0.0	0.0
SN123M	8.10E 02	6.86E 02	4.92E 02	3.53E 02	1.86E 00	0.0
TE123M	3.48E 00	2.92E 00	2.04E 00	1.43E 00	5.27E-03	0.0
SB124	1.09E 02	7.67E 01	3.84E 01	1.92E 01	3.46E-04	0.0
SN125	6.11E 04	6.69E 03	8.02E 01	9.61E-01	0.0	0.0
SB125	2.01E 04	2.02E 04	1.94E 04	1.86E 04	9.56E 03	9.32E 00
TE125M	6.74E 03	7.08E 03	7.40E 03	7.42E 03	3.96E 03	3.86E 00
SN126	1.70E 00	1.70E 00	1.70E 00	1.70E 00	1.70E 00	1.70E 00
SB126M	3.07E 03	1.70E 00	1.70E 00	1.70E 00	1.70E 00	1.70E 00
SB126	4.99E 03	9.47E 02	3.56E 01	2.90E 00	1.68E 00	1.68E 00
SB127	3.36E 05	1.60E 03	3.52E-02	7.74E-07	0.0	0.0
TE127M	7.11E 04	6.11E 04	4.17E 04	2.85E 04	6.95E 01	0.0
TE127	3.32E 05	6.18E 04	4.12E 04	2.81E 04	6.87E 01	6.00E-06
TE129M	3.32E 05	1.81E 05	5.33E 04	1.57E 04	6.61E-05	0.0

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
TE129	8.02E 05	1.16E 05	3.42E 04	1.01E 04	4.23E-05	0.0
I129	5.24E-02	5.33E-02	5.40E-02	5.42E-02	5.43E-02	5.43E-02
I131	1.79E 06	1.39E 05	7.92E 02	4.52E 00	0.0	0.0
XE131M	1.84E 04	6.19E 03	2.39E 02	7.37E 00	0.0	0.0
TE132	2.50E 06	4.18E 03	1.19E-02	3.39E-08	0.0	0.0
I132	2.52E 06	4.30E 03	1.23E-02	3.49E-08	0.0	0.0
XE133M	8.06E 04	1.22E 01	1.24E-07	1.27E-15	0.0	0.0
XE133	3.26E 06	7.45E 04	2.79E 01	1.04E-02	0.0	0.0
CS134	2.98E 04	2.90E 04	2.74E 04	2.59E 04	1.08E 04	1.20E 00
CS135	1.17E 00	1.17E 00	1.17E 00	1.17E 00	1.17E 00	1.17E 00
CS136	1.42E 05	2.88E 04	1.17E 03	4.79E 01	0.0	0.0
CS137	1.09E 05	1.09E 05	1.09E 05	1.08E 05	1.02E 05	5.46E 04
BA137M	1.02E 05	1.02E 05	1.02E 05	1.01E 05	9.53E 04	5.11E 04
BA140	2.65E 06	5.23E 05	2.03E 04	7.87E 02	0.0	0.0
LA140	2.75E 06	6.01E 05	2.33E 04	9.06E 02	9.05E-16	0.0
CE141	2.80E 06	1.48E 06	4.10E 05	1.14E 05	1.85E-04	0.0
CE143	2.64E 06	7.18E-01	5.25E-14	0.0	0.0	0.0
PR143	2.64E 06	6.44E 05	3.09E 04	1.49E 03	2.43E-18	0.0
CE144	1.37E 06	1.28E 06	1.10E 06	9.53E 05	9.47E 04	3.33E-06
PR144	1.43E 06	1.28E 06	1.10E 06	9.53E 05	9.47E 04	3.33E-06
ND147	1.20E 06	1.85E 05	4.37E 03	1.03E 02	0.0	0.0
PM147	3.49E 05	3.53E 05	3.40E 05	3.26E 05	1.64E 05	1.30E 02
PM148M	6.80E 04	4.14E 04	1.54E 04	5.72E 03	9.48E-04	0.0
PM148	8.06E 04	4.93E 03	1.24E 03	4.59E 02	7.62E-05	0.0
PM149	7.17E 05	6.15E 01	4.22E-07	2.90E-15	0.0	0.0
SM151	4.68E 03	4.69E 03	4.69E 03	4.68E 03	4.59E 03	3.70E 03
EU152	1.05E 01	1.05E 01	1.04E 01	1.03E 01	8.84E 00	1.86E 00
SM153	1.95E 05	4.82E 00	2.89E-09	0.0	0.0	0.0
GD153	5.49E-01	5.04E-01	4.24E-01	3.57E-01	2.38E-02	1.28E-14
EU154	9.79E 02	9.76E 02	9.69E 02	9.62E 02	8.60E 02	2.67E 02
EU155	8.19E 04	7.94E 04	7.46E 04	7.00E 04	2.60E 04	8.39E-01
EU156	1.20E 05	3.06E 04	1.91E 03	1.20E 02	1.24E-17	0.0
TB160	1.26E 04	9.45E 03	5.31E 03	2.98E 03	3.35E-01	0.0
TB161	1.85E 04	9.08E 02	2.19E 00	5.28E-03	0.0	0.0
GD162	4.68E 03	4.42E 03	3.94E 03	3.52E 03	5.84E 02	4.35E-06
TB162M	4.68E 03	4.42E 03	3.94E 03	3.52E 03	5.84E 02	4.35E-06
DY166	2.80E 02	6.13E-01	2.98E-06	1.45E-11	0.0	0.0
HO166	3.71E 02	9.14E-01	4.45E-06	2.16E-11	0.0	0.0
SUBTOT	5.66E 07	2.01E 07	1.09E 07	7.41E 06	1.04E 06	1.53E 05
TOTALS	2.59E 08	2.01E 07	1.09E 07	7.41E 06	1.04E 06	1.53E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H 3	3.33E-02	3.31E-02	3.28E-02	3.25E-02	2.81E-02	6.14E-03
KR 85	1.65E 01	1.64E 01	1.62E 01	1.61E 01	1.35E 01	2.39E 00
RB 86	1.47E 01	4.82E 00	5.19E-01	5.59E-02	3.16E-17	0.0
SR 89	3.42E 03	2.29E 03	1.03E 03	4.63E 02	1.55E-03	0.0
SR 90	5.68E 01	5.67E 01	5.64E 01	5.62E 01	5.27E 01	2.71E 01
Y 90	2.57E 02	2.49E 02	2.48E 02	2.47E 02	2.31E 02	1.19E 02
Y 91	4.98E 03	3.50E 03	1.73E 03	8.52E 02	1.23E-02	0.0
ZR 95	1.51E 04	1.10E 04	5.79E 03	3.05E 03	1.28E-01	0.0
NB 95M	8.04E 01	6.20E 01	3.28E 01	1.73E 01	7.21E-04	1.05E-15
NB 95	1.37E 04	1.28E 04	8.75E 03	5.23E 03	2.53E-01	0.0
MO 99	1.39E 04	8.12E 00	2.75E-06	9.32E-13	0.0	0.0
TC 99M	2.29E 03	1.46E 00	4.96E-07	1.68E-13	0.0	0.0
RU103	9.71E 03	5.75E 03	2.01E 03	7.03E 02	4.53E-05	0.0
RH103M	7.05E 02	4.18E 02	1.46E 02	5.11E 01	3.29E-06	0.0
RU106	8.09E 01	7.65E 01	6.83E 01	6.10E 01	1.02E 01	8.32E-08
RH106	1.32E 04	1.25E 04	1.11E 04	9.94E 03	1.66E 03	1.36E-05
AG110M	2.78E 01	2.56E 01	2.17E 01	1.84E 01	1.38E 00	2.54E-12
AG110	8.67E 02	1.50E 00	1.27E 00	1.08E 00	8.06E-02	1.48E-13
AG111	4.82E 02	3.01E 01	1.18E-01	4.60E-04	0.0	0.0
CD113M	1.67E-01	1.67E-01	1.65E-01	1.64E-01	1.44E-01	3.79E-02
CD115M	1.78E 00	1.10E 00	4.17E-01	1.59E-01	3.78E-08	0.0
SN121M	5.68E-02	5.67E-02	5.66E-02	5.65E-02	5.52E-02	4.32E-02
SN123M	2.76E 00	2.34E 00	1.68E 00	1.20E 00	6.33E-03	0.0
SB124	1.47E 00	1.04E 00	5.19E-01	2.59E-01	4.68E-06	0.0
SN125	3.70E 02	4.05E 01	4.85E-01	5.82E-03	0.0	0.0
SB125	6.74E 01	6.77E 01	6.51E 01	6.24E 01	3.21E 01	3.13E-02
TE125M	5.79E 00	6.09E 00	6.36E 00	6.38E 00	3.41E 00	3.32E-03
SB126	5.47E 01	1.04E 01	3.90E-01	3.18E-02	1.84E-02	1.84E-02
SB127	1.85E 03	8.81E 00	1.94E-04	4.25E-09	0.0	0.0
TE127M	3.92E 01	3.37E 01	2.30E 01	1.57E 01	3.83E-02	0.0
TE127	5.41E 02	1.01E 02	6.72E 01	4.59E 01	1.12E-01	9.78E-09
TE129M	6.58E 02	3.59E 02	1.06E 02	3.11E 01	1.31E-07	0.0
TE129	2.91E 03	4.21E 02	1.24E 02	3.65E 01	1.54E-07	0.0
I131	7.35E 03	5.71E 02	3.26E 00	1.86E-02	0.0	0.0
XE131M	1.79E 01	6.02E 00	2.32E-01	7.16E-03	0.0	0.0
TE132	3.42E 03	5.69E 00	1.62E-05	4.62E-11	0.0	0.0
I132	4.03E 04	6.89E 01	1.96E-04	5.59E-10	0.0	0.0
XE133	3.52E 03	8.04E 01	3.01E-02	1.12E-05	0.0	0.0
CS134	3.12E 02	3.04E 02	2.87E 02	2.72E 02	1.13E 02	1.26E-02
CS136	2.20E 03	4.45E 02	1.82E 01	7.40E-01	0.0	0.0
CS137	1.77E 02	1.77E 02	1.76E 02	1.75E 02	1.65E 02	8.84E 01
BA137M	4.01E 02	4.00E 02	3.99E 02	3.97E 02	3.74E 02	2.00E 02
BA140	8.95E 03	1.76E 03	6.84E 01	2.65E 00	0.0	0.0
LA140	4.56E 04	9.98E 03	3.87E 02	1.50E 01	1.50E-17	0.0
CE141	5.51E 03	2.91E 03	8.07E 02	2.24E 02	3.63E-07	0.0
PR143	5.73E 03	1.40E 03	6.71E 01	3.22E 00	5.27E-21	0.0
CE144	1.21E 03	1.12E 03	9.68E 02	8.36E 02	8.31E 01	2.93E-09

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
PR144	1.07E 04	9.50E 03	8.21E 03	7.09E 03	7.05E 02	2.48E-08
ND147	3.37E 03	5.18E 02	1.22E 01	2.88E-01	0.0	0.0
PM147	1.80E 02	1.82E 02	1.75E 02	1.68E 02	8.46E 01	6.68E-02
PM148M	8.63E 02	5.26E 02	1.95E 02	7.26E 01	1.20E-05	0.0
PM148	6.59E 02	4.03E 01	1.01E 01	3.76E 00	6.23E-07	0.0
PM149	1.79E 03	1.54E-01	1.05E-09	7.23E-18	0.0	0.0
SM151	8.16E 00	8.18E 00	8.17E 00	8.16E 00	7.99E 00	6.44E 00
EU152	1.28E-01	1.28E-01	1.27E-01	1.25E-01	1.08E-01	2.27E-02
EU154	9.17E 00	9.14E 00	9.08E 00	9.01E 00	8.05E 00	2.50E 00
EU155	6.90E 01	6.68E 01	6.28E 01	5.89E 01	2.19E 01	7.06E-04
EU156	1.27E 03	3.23E 02	2.02E 01	1.26E 00	1.31E-19	0.0
TB160	1.06E 02	7.96E 01	4.47E 01	2.51E 01	2.82E-03	0.0
TB161	3.01E 01	1.48E 00	3.57E-03	8.61E-06	0.0	0.0
GD162	1.59E 01	1.50E 01	1.34E 01	1.20E 01	1.99E 00	1.48E-08
TB162M	3.13E 01	2.96E 01	2.64E 01	2.36E 01	3.91E 00	2.91E-08
SUBTOT	2.29E 05	8.03E 04	4.34E 04	3.03E 04	3.58E 03	4.46E 02
TOTALS	2.65E 06	8.03E 04	4.34E 04	3.03E 04	3.58E 03	4.46E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1095. D	10958. D
KR 85	1.32E-01	1.31E-01	1.30E-01	1.28E-01	1.09E-01	1.91E-02
RB 86	1.70E 00	5.59E-01	6.02E-02	6.48E-03	3.67E-18	0.0
Y 91	2.99E 01	2.10E 01	1.04E 01	5.11E 00	7.37E-05	0.0
ZR 95	1.24E 04	9.01E 03	4.75E 03	2.51E 03	1.05E-01	0.0
NB 95M	8.04E 01	5.20E 01	3.28E 01	1.73E 01	7.21E-04	1.05E-15
NB 95	1.29E 04	1.21E 04	8.27E 03	4.94E 03	2.39E-01	0.0
MO 99	4.04E 03	2.35E 00	7.98E-07	2.70E-13	0.0	0.0
RU103	8.55E 03	5.06E 03	1.77E 03	6.19E 02	3.98E-05	0.0
RH106	1.58E 03	1.50E 03	1.34E 03	1.19E 03	2.00E 02	1.63E-06
AG110M	2.67E 01	2.45E 01	2.08E 01	1.77E 01	1.32E 00	2.44E-12
AG110	2.34E 01	4.04E-02	3.43E-02	2.91E-02	2.18E-03	4.01E-15
CD115M	2.61E-01	1.61E-01	6.13E-02	2.33E-02	5.56E-09	0.0
SB124	1.20E 00	3.50E-01	4.25E-01	2.13E-01	3.83E-06	0.0
SN125	3.37E 01	3.69E 00	4.42E-02	5.29E-04	0.0	0.0
SB125	3.71E 01	3.72E 01	3.58E 01	3.43E 01	1.76E 01	1.72E-02
SB126	3.17E 01	6.02E 00	2.26E-01	1.84E-02	1.07E-02	1.07E-02
SB127	9.35E 02	4.47E 00	9.82E-05	2.16E-09	0.0	0.0
TE127	8.12E 00	1.51E 00	1.01E 00	6.88E-01	1.63E-03	1.47E-10
TE129M	7.89E 01	4.30E 01	1.27E 01	3.73E 00	1.57E-08	0.0
TE129	4.36E 02	6.32E 01	1.86E 01	5.47E 00	2.30E-08	0.0
I131	4.22E 03	3.28E 02	1.87E 00	1.07E-02	0.0	0.0
TE132	1.30E 03	2.16E 00	6.16E-06	1.76E-11	0.0	0.0
I132	3.14E 04	5.37E 01	1.53E-04	4.36E-10	0.0	0.0
CS134	2.78E 02	2.70E 02	2.56E 02	2.42E 02	1.01E 02	1.12E-02
CS136	2.07E 03	4.18E 02	1.71E 01	6.96E-01	0.0	0.0
BA137M	4.01E 02	4.00E 02	3.99E 02	3.97E 02	3.74E 02	2.00E 02
BA140	4.21E 03	8.28E 02	3.22E 01	1.25E 00	0.0	0.0
LA140	3.51E 04	7.68E 03	2.98E 02	1.16E 01	1.16E-17	0.0
CE144	3.26E 02	3.03E 02	2.61E 02	2.26E 02	2.24E 01	7.90E-10
PR144	2.56E 02	2.28E 02	1.97E 02	1.70E 02	1.69E 01	5.95E-10
ND147	1.45E 02	2.23E 01	5.26E-01	1.24E-02	0.0	0.0
PM148M	7.85E 02	4.79E 02	1.78E 02	6.61E 01	1.10E-05	0.0
PM148	3.03E 02	1.85E 01	4.66E 00	1.73E 00	2.87E-07	0.0
EU154	7.71E 00	7.68E 00	7.62E 00	7.57E 00	6.77E 00	2.10E 00
EU156	9.62E 02	2.46E 02	1.54E 01	9.60E-01	9.93E-20	0.0
TB160	8.71E 01	6.52E 01	3.66E 01	2.06E 01	2.31E-03	0.0
SUBTOT	1.23E 05	3.93E 04	1.80E 04	1.05E 04	7.40E 02	2.03E 02
TOTALS	9.22E 05	3.93E 04	1.80E 04	1.05E 04	7.40E 02	2.03E 02

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
H	3.33E-02	3.31E-02	3.28E-02	3.25E-02	2.81E-02	6.14E-03
ZN	7.53E-01	1.64E-05	7.83E-15	0.0	0.0	0.0
GA	1.59E 01	3.32E-04	1.58E-13	0.0	0.0	0.0
GE	6.93E 00	1.00E-19	0.0	0.0	0.0	0.0
AS	1.21E 02	1.24E-06	7.83E-18	0.0	0.0	0.0
SE	5.85E 03	1.49E-05	1.49E-05	1.49E-05	1.49E-05	1.49E-05
BR	5.62E 04	3.40E-06	1.84E-18	0.0	0.0	0.0
KR	8.13E 04	1.64E 01	1.62E 01	1.61E 01	1.36E 01	2.39E 00
RB	2.20E 05	4.82E 00	5.19E-01	5.59E-02	7.18E-09	7.18E-09
SR	1.37E 05	2.35E 03	1.09E 03	5.19E 02	5.27E 01	2.71E 01
Y	2.00E 05	3.75E 03	1.98E 03	1.10E 03	2.31E 02	1.19E 02
ZR	3.48E 04	1.10E 04	5.79E 03	3.05E 03	1.28E-01	1.69E-04
NB	1.76E 05	1.28E 04	8.78E 03	5.25E 03	2.54E-01	2.01E-04
MO	1.13E 05	8.12E 00	2.75E-06	9.32E-13	0.0	0.0
TC	1.69E 05	1.47E 00	1.00E-02	1.00E-02	1.00E-02	1.00E-02
RU	4.07E 04	5.82E 03	2.08E 03	7.64E 02	1.02E 01	8.32E-08
RH	4.92E 04	1.29E 04	1.13E 04	9.99E 03	1.66E 03	1.36E-05
PD	4.77E 03	2.86E-09	0.0	0.0	0.0	0.0
AG	4.00E 03	5.72E 01	2.31E 01	1.95E 01	1.46E 00	2.69E-12
CD	3.02E 02	1.27E 00	5.82E-01	3.22E-01	1.44E-01	3.79E-02
IN	6.47E 02	9.70E-03	3.71E-03	1.61E-03	3.25E-09	0.0
SN	1.39E 04	4.29E 01	2.23E 00	1.27E 00	6.30E-02	4.41E-02
SB	1.66E 05	8.79E 01	6.60E 01	6.27E 01	3.21E 01	5.07E-02
TE	1.26E 05	9.26E 02	3.26E 02	1.35E 02	3.55E 00	3.32E-03
I	2.59E 05	6.40E 02	3.26E 00	1.86E-02	2.35E-05	2.35E-05
XE	1.56E 05	8.64E 01	2.62E-01	7.17E-03	0.0	0.0
CS	2.56E 05	9.25E 02	4.81E 02	4.48E 02	2.78E 02	8.84E 01
BA	8.85E 04	2.16E 03	4.67E 02	4.00E 02	3.74E 02	2.00E 02
LA	1.60E 05	9.98E 03	3.87E 02	1.50E 01	1.50E-17	0.0
CE	4.96E 04	4.03E 03	1.78E 03	1.06E 03	8.31E 01	2.93E-09
PR	6.34E 04	1.09E 04	8.28E 03	7.09E 03	7.05E 02	2.48E-08
ND	1.02E 04	5.18E 02	1.22E 01	2.88E-01	0.0	0.0
PM	8.66E 03	7.49E 02	3.81E 02	2.44E 02	8.46E 01	6.68E-02
SM	1.45E 03	3.19E 00	8.17E 00	8.16E 00	7.99E 00	6.44E 00
EU	4.08E 03	3.99E 02	9.22E 01	6.93E 01	3.00E 01	2.52E 00
GD	3.49E 02	1.50E 01	1.34E 01	1.20E 01	1.99E 00	1.48E-08
TB	1.76E 02	1.11E 02	7.11E 01	4.87E 01	3.91E 00	2.91E-08
DY	2.05E 00	5.19E-04	2.53E-09	1.23E-14	0.0	0.0
HO	1.62E 00	4.01E-03	1.95E-08	9.47E-14	0.0	0.0
TOTALS	2.65E 06	3.03E 04	4.34E 04	3.03E 04	3.58E 03	4.46E 02

APPENDIX B: CONCENTRATIONS, RADIOACTIVITY, AND THERMAL POWER OF ACTINIDES IN THE CORE, AXIAL BLANKET, AND RADIAL BLANKET OF THE ATOMICS INTERNATIONAL REFERENCE OXIDE LMFBR

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

CONCENTRATION HISTORY OF ACTINIDES DURING REACTOR LIFE

NUCLIDE CONCENTRATIONS, GRAM ATOMS / METRIC TON (U + PU)

	CHARGE	108. D	216. D	324. D	432. D	540. D
20040	0.0	3.90E-02	9.36E-02	1.73E-01	2.80E-01	4.16E-01
812080	0.0	4.28E-17	3.39E-16	1.13E-15	2.66E-15	5.14E-15
822080	0.0	1.04E-12	1.66E-11	8.32E-11	2.61E-10	6.32E-10
822120	0.0	2.45E-14	1.94E-13	6.49E-13	1.52E-12	2.94E-12
832120	0.0	2.33E-15	1.84E-14	6.16E-14	1.45E-13	2.79E-13
842120	0.0	4.09E-23	3.24E-22	1.08E-21	2.54E-21	4.91E-21
842160	0.0	9.59E-20	7.60E-19	2.54E-18	5.96E-18	1.15E-17
862200	0.0	3.58E-17	2.84E-16	9.49E-16	2.23E-15	4.29E-15
882240	0.0	2.01E-13	1.59E-12	5.33E-12	1.25E-11	2.41E-11
902280	0.0	3.86E-11	3.06E-10	1.02E-09	2.40E-09	4.63E-09
902290	0.0	2.83E-13	3.51E-12	1.60E-11	4.74E-11	1.11E-10
902300	0.0	9.90E-09	3.76E-08	8.03E-08	1.35E-07	2.00E-07
902310	0.0	2.51E-12	9.92E-12	2.20E-11	3.83E-11	5.85E-11
902320	0.0	3.44E-11	1.35E-10	2.97E-10	5.18E-10	7.91E-10
902330	0.0	1.40E-16	5.69E-16	1.30E-15	2.35E-15	3.71E-15
912310	0.0	5.88E-11	4.61E-10	1.52E-09	3.50E-09	6.63E-09
912320	0.0	3.69E-14	3.01E-13	1.03E-12	2.45E-12	4.80E-12
912330	0.0	1.29E-13	7.07E-13	1.81E-12	3.46E-12	5.67E-12
912341	0.0	2.55E-20	1.46E-19	3.87E-19	7.67E-19	1.30E-18
912340	0.0	8.43E-18	4.81E-17	1.28E-16	2.53E-16	4.29E-16
922320	0.0	4.24E-08	1.76E-07	4.09E-07	7.48E-07	1.20E-06
922330	0.0	2.77E-07	1.04E-06	2.18E-06	3.60E-06	5.22E-06
922340	0.0	2.37E-02	4.47E-02	6.31E-02	7.92E-02	9.31E-02
922350	0.0	4.71E-03	9.73E-03	1.50E-02	2.05E-02	2.63E-02
922360	0.0	8.01E-03	1.57E-02	2.31E-02	3.02E-02	3.69E-02
922370	0.0	1.44E-02	1.48E-02	1.51E-02	1.54E-02	1.56E-02
922380	3.28E 03	3.23E 03	3.18E 03	3.13E 03	3.07E 03	3.02E 03
922390	0.0	9.47E-03	9.68E-03	9.87E-03	1.00E-02	1.02E-02
932360	0.0	8.38E-09	1.72E-08	2.64E-08	3.59E-08	4.55E-08
932370	0.0	1.58E-01	3.11E-01	4.60E-01	6.04E-01	7.42E-01
932380	0.0	1.50E-04	3.08E-04	4.73E-04	6.43E-04	8.16E-04
932390	0.0	1.36E 00	1.39E 00	1.42E 00	1.44E 00	1.47E 00
942360	0.0	1.17E-06	2.53E-06	4.05E-06	5.73E-06	7.54E-06
942380	1.09E 01	1.01E 01	9.43E 00	8.77E 00	8.18E 00	7.65E 00
942390	5.42E 02	5.31E 02	5.19E 02	5.08E 02	4.97E 02	4.86E 02
942400	2.16E 02	2.17E 02	2.18E 02	2.18E 02	2.18E 02	2.18E 02
942410	1.08E 02	9.54E 01	8.45E 01	7.51E 01	6.71E 01	6.04E 01
942420	3.62E 01	3.67E 01	3.71E 01	3.73E 01	3.73E 01	3.73E 01
942430	0.0	1.63E-03	1.71E-03	1.79E-03	1.85E-03	1.91E-03
952410	0.0	1.55E 00	2.81E 00	3.83E 00	4.64E 00	5.27E 00
952421	0.0	5.53E-03	2.06E-02	4.30E-02	7.06E-02	1.02E-01
952420	0.0	5.10E-04	9.63E-04	1.36E-03	1.71E-03	2.01E-03
952430	0.0	5.76E-01	1.16E 00	1.75E 00	2.34E 00	2.92E 00
952440	0.0	3.44E-06	7.20E-06	1.13E-05	1.56E-05	2.01E-05
962420	0.0	2.06E-02	6.86E-02	1.29E-01	1.93E-01	2.55E-01
962430	0.0	1.22E-04	8.68E-04	2.60E-03	5.48E-03	9.55E-03
962440	0.0	6.96E-03	2.79E-02	6.29E-02	1.12E-01	1.73E-01
FLUX		4.78E 15	4.97E 15	5.15E 15	5.34E 15	5.52E 15

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U234	0.0	2.18E 01	2.29E 01	2.52E 01	2.76E 01	6.43E 01	4.07E 02
U238	7.82E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05
NP237	0.0	1.76E 02	1.80E 02	1.80E 02	1.80E 02	1.90E 02	5.55E 02
PU238	2.59E 03	1.82E 03	1.83E 03	1.84E 03	1.84E 03	1.84E 03	1.49E 03
PU239	1.30E 05	1.16E 05	1.17E 05	1.17E 05	1.17E 05	1.17E 05	1.16E 05
PU240	5.18E 04	5.24E 04	5.24E 04	5.24E 04	5.24E 04	5.24E 04	5.23E 04
PU241	2.60E 04	1.45E 04	1.45E 04	1.44E 04	1.42E 04	1.24E 04	2.94E 03
PU242	8.76E 03	9.02E 03	9.02E 03	9.02E 03	9.02E 03	9.02E 03	9.02E 03
AM241	0.0	1.27E 03	1.33E 03	1.46E 03	1.59E 03	3.41E 03	1.25E 04
AM242M	0.0	2.46E 01	2.46E 01	2.46E 01	2.46E 01	2.43E 01	2.15E 01
AM243	0.0	7.10E 02	7.11E 02	7.11E 02	7.11E 02	7.10E 02	7.09E 02
CM242	0.0	6.16E 01	5.46E 01	4.23E 01	3.28E 01	6.45E-01	5.17E-02
CM244	0.0	4.23E 01	4.22E 01	4.19E 01	4.17E 01	3.77E 01	1.34E 01
SUBTOT	1.00E 06	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05
TOTALS	1.00E 06	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
HE	0.0	1.66E 00	1.82E 00	2.10E 00	2.33E 00	4.11E 00	2.05E 01
TL	0.0	1.07E-12	1.21E-12	1.68E-12	2.21E-12	1.54E-11	4.16E-11
PB	0.0	1.32E-07	1.63E-07	2.41E-07	3.45E-07	7.47E-06	3.81E-04
BI	0.0	5.91E-11	6.69E-11	9.30E-11	1.22E-10	8.50E-10	2.30E-09
PO	0.0	2.49E-15	2.87E-15	3.91E-15	5.13E-15	3.57E-14	9.69E-14
RN	0.0	9.45E-13	1.09E-12	1.49E-12	1.95E-12	1.36E-11	3.68E-11
RA	0.0	5.40E-09	6.23E-09	8.50E-09	1.12E-08	7.77E-08	2.11E-07
TH	0.0	4.73E-05	5.25E-05	6.40E-05	7.65E-05	4.19E-04	1.85E-02
PA	0.0	1.53E-06	1.55E-06	1.55E-06	1.54E-06	1.55E-06	1.56E-06
U	7.82E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05	7.19E 05
NP	0.0	5.26E 02	1.80E 02	1.80E 02	1.80E 02	1.90E 02	5.55E 02
PU	2.19E 05	1.94E 05	1.94E 05	1.94E 05	1.94E 05	1.92E 05	1.82E 05
AM	0.0	2.01E 03	2.07E 03	2.20E 03	2.32E 03	4.15E 03	1.32E 04
CM	0.0	1.06E 02	9.91E 01	8.66E 01	7.68E 01	4.06E 01	1.47E 01
TOTALS	1.00E 06	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05	9.15E 05

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U234	0.0	1.35E-01	1.42E-01	1.56E-01	1.71E-01	3.98E-01	2.52E 00
U237	0.0	3.02E 05	1.39E 04	2.92E 01	6.17E-02	0.0	0.0
U238	2.60E-01	2.39E-01	2.39E-01	2.39E-01	2.39E-01	2.39E-01	2.39E-01
NP237	0.0	1.24E-01	1.27E-01	1.27E-01	1.27E-01	1.34E-01	3.92E-01
NP238	0.0	5.07E 04	2.54E 00	6.38E-09	1.60E-17	0.0	0.0
NP239	0.0	8.15E 07	1.19E 04	1.37E 02	1.37E 02	1.37E 02	1.36E 02
PU236	0.0	9.46E-01	9.31E-01	8.94E-01	8.59E-01	4.58E-01	6.44E-04
PU238	4.38E 04	3.07E 04	3.08E 04	3.10E 04	3.11E 04	3.10E 04	2.52E 04
PU239	7.94E 03	7.12E 03	7.15E 03	7.15E 03	7.15E 03	7.15E 03	7.14E 03
PU240	1.14E 04	1.16E 04	1.16E 04	1.16E 04	1.16E 04	1.15E 04	1.15E 04
PU241	2.97E 06	1.66E 06	1.65E 06	1.64E 06	1.62E 06	1.42E 06	3.35E 05
PU242	3.42E 01	3.52E 01	3.52E 01	3.52E 01	3.52E 01	3.52E 01	3.52E 01
AM241	0.0	4.12E 03	4.32E 03	4.73E 03	5.14E 03	1.11E 04	4.05E 04
AM242M	0.0	2.39E 02	2.39E 02	2.39E 02	2.39E 02	2.36E 02	2.09E 02
AM242	0.0	3.93E 05	2.39E 02	2.39E 02	2.39E 02	2.36E 02	2.09E 02
AM243	0.0	1.37E 02	1.37E 02	1.37E 02	1.37E 02	1.37E 02	1.36E 02
CM242	0.0	2.04E 05	1.81E 05	1.40E 05	1.09E 05	2.13E 03	1.71E 02
CM243	0.0	1.07E 02	1.07E 02	1.06E 02	1.06E 02	1.00E 02	5.57E 01
CM244	0.0	3.43E 03	3.42E 03	3.40E 03	3.38E 03	3.06E 03	1.09E 03
SUBTOT	3.03E 06	8.42E 07	1.92E 06	1.84E 06	1.79E 06	1.48E 06	4.22E 05
TOTALS	3.03E C6	1.67E 08	1.92E 06	1.84E 06	1.79E 06	1.48E 06	4.22E 05

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	6.48E 02	2.98E 01	6.28E-02	1.32E-04	0.0	0.0
NP239	0.0	2.42E 05	3.53E 01	4.05E-01	4.05E-01	4.05E-01	4.04E-01
PU238	1.45E 03	1.02E 03	1.02E 03	1.03E 03	1.03E 03	1.03E 03	8.34E 02
PU239	2.47E 02	2.21E 02	2.22E 02	2.22E 02	2.22E 02	2.22E 02	2.22E 02
PU240	3.56E 02	3.60E 02	3.60E 02	3.60E 02	3.60E 02	3.60E 02	3.59E 02
PU241	1.23E 02	6.89E 01	6.86E 01	6.80E 01	6.74E 01	5.87E 01	1.39E 01
PU242	1.01E 00	1.04E 00	1.04E 00	1.04E 00	1.04E 00	1.04E 00	1.04E 00
AM241	0.0	1.37E 02	1.44E 02	1.58E 02	1.71E 02	3.69E 02	1.35E 03
AM243	0.0	4.40E-01	4.40E-01	4.40E-01	4.40E-01	4.40E-01	4.39E-01
CM242	0.0	7.51E 02	6.65E 02	5.16E 02	4.00E 02	7.86E 00	6.30E-01
CM243	0.0	3.90E-01	3.89E-01	3.88E-01	3.86E-01	3.65E-01	2.04E-01
CM244	0.0	1.20E 01	1.20E 01	1.19E 01	1.18E 01	1.07E 01	3.80E 00
SUBTOT	2.18E 03	2.45E 05	2.56E 03	2.36E 03	2.27E 03	2.06E 03	2.79E 03
TOTALS	2.18E 03	4.41E 05	2.56E 03	2.36E 03	2.27E 03	2.06E 03	2.79E 03

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	4.60E 02	2.11E 01	4.46E-02	9.40E-05	0.0	0.0
NP239	0.0	1.57E 05	2.30E 01	2.63E-01	2.63E-01	2.63E-01	2.63E-01
AM243	0.0	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.96E-01	3.95E-01
CM242	0.0	5.26E 02	4.66E 02	3.61E 02	2.80E 02	5.50E 00	4.41E-01
CM244	0.0	2.40E 00	2.39E 00	2.38E 00	2.35E 00	2.14E 00	7.60E-01
SUBTOT	0.0	1.58E 05	5.13E 02	3.64E 02	2.83E 02	8.30E 00	1.86E 00
TOTALS	0.0	1.58E 05	5.13E 02	3.64E 02	2.83E 02	8.35E 00	1.89E 00

AI REFERENCE OXIDE LMFBR CORE
POWER= 148.15 MW/MT, BURNUP= 80001. MWD/MT, FLUX= 5.15E 15 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TL	0.0	6.09E-06	6.90E-06	9.59E-06	1.26E-05	8.77E-05	2.38E-04
PB	0.0	1.53E-06	1.74E-06	2.41E-06	3.17E-06	2.21E-05	5.98E-05
BI	0.0	1.39E-05	1.57E-05	2.19E-05	2.87E-05	2.00E-04	5.42E-04
PO	0.0	6.47E-05	7.41E-05	1.02E-04	1.34E-04	9.31E-04	2.52E-03
RN	0.0	3.28E-05	3.79E-05	5.16E-05	6.78E-05	4.72E-04	1.28E-03
RA	0.0	2.97E-05	3.42E-05	4.67E-05	6.12E-05	4.27E-04	1.16E-03
TH	0.0	3.84E-05	3.37E-05	4.58E-05	5.98E-05	4.08E-04	1.11E-03
PA	0.0	1.65E-06	3.44E-08	9.29E-09	3.79E-09	2.25E-09	2.27E-09
U	6.59E-03	1.94E 05	2.98E 01	7.36E-02	1.14E-02	1.84E-02	8.00E-02
NP	0.0	2.42E 05	3.53E 01	4.05E-01	4.05E-01	4.05E-01	4.04E-01
PU	2.18E 03	3.38E 03	1.67E 03	1.68E 03	1.68E 03	1.67E 03	1.43E 03
AM	0.0	6.52E 02	1.45E 02	1.58E 02	1.72E 02	3.69E 02	1.35E 03
CM	0.0	7.63E 02	6.78E 02	5.28E 02	4.12E 02	1.89E 01	4.63E 00
TOTALS	2.18E 03	4.41E 05	2.56E 03	2.36E 03	2.27E 03	2.06E 03	2.79E 03

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

CONCENTRATION HISTORY OF ACTINIDES DURING REACTOR LIFE

NUCLIDE CONCENTRATIONS, GRAM ATOMS / METRIC TON (U + PU)

	CHARGE	108. D	216. D	324. D	432. D	540. D
20040	0.0	3.21E-05	1.38E-04	3.35E-04	6.36E-04	1.05E-03
812080	0.0	8.71E-20	1.49E-18	8.41E-18	2.92E-17	7.68E-17
822080	0.0	1.71E-15	5.62E-14	4.66E-13	2.13E-12	7.00E-12
822120	0.0	4.98E-17	8.55E-16	4.81E-15	1.67E-14	4.39E-14
832120	0.0	4.73E-18	8.12E-17	4.57E-16	1.58E-15	4.17E-15
842120	0.0	8.33E-26	1.43E-24	8.04E-24	2.79E-23	7.34E-23
842160	0.0	1.95E-22	3.35E-21	1.88E-20	6.53E-20	1.72E-19
862200	0.0	7.28E-20	1.25E-18	7.03E-18	2.44E-17	6.42E-17
882240	0.0	4.09E-16	7.02E-15	3.95E-14	1.37E-13	3.61E-13
902280	0.0	7.84E-14	1.35E-12	7.57E-12	2.63E-11	6.92E-11
902290	0.0	1.61E-13	6.84E-13	1.64E-12	3.09E-12	5.14E-12
902300	0.0	3.40E-11	1.46E-10	3.54E-10	6.70E-10	1.11E-09
902310	0.0	5.02E-13	5.06E-13	5.17E-13	5.37E-13	5.67E-13
902320	0.0	9.29E-11	3.98E-10	9.64E-10	1.82E-09	3.00E-09
902330	0.0	5.10E-17	2.86E-16	8.22E-16	1.75E-15	3.13E-15
912310	0.0	3.51E-11	6.99E-11	1.05E-10	1.40E-10	1.77E-10
912320	0.0	2.98E-15	7.79E-15	1.39E-14	2.08E-14	2.86E-14
912330	0.0	4.72E-14	3.50E-13	1.12E-12	2.54E-12	4.72E-12
912341	0.0	1.26E-21	1.23E-20	4.68E-20	1.19E-19	2.40E-19
912340	0.0	4.17E-19	4.06E-18	1.55E-17	3.92E-17	7.93E-17
922320	0.0	1.12E-10	1.05E-09	4.12E-09	1.11E-08	2.39E-08
922330	0.0	2.56E-07	5.83E-07	9.58E-07	1.37E-06	1.79E-06
922340	0.0	8.21E-05	1.89E-04	3.16E-04	4.58E-04	6.13E-04
922350	1.22E 01	1.22E 01	1.21E 01	1.21E 01	1.21E 01	1.20E 01
922360	0.0	2.17E-02	4.97E-02	8.27E-02	1.19E-01	1.58E-01
922370	0.0	2.55E-03	3.34E-03	3.96E-03	4.45E-03	4.84E-03
922380	4.19E 03	4.18E 03	4.17E 03	4.16E 03	4.14E 03	4.12E 03
922390	0.0	1.65E-03	2.16E-03	2.56E-03	2.86E-03	3.10E-03
932360	0.0	2.03E-10	6.10E-10	1.21E-09	1.96E-09	2.83E-09
932370	0.0	2.82E-02	6.47E-02	1.08E-01	1.56E-01	2.07E-01
932380	0.0	3.64E-06	1.10E-05	2.17E-05	3.51E-05	5.08E-05
932390	0.0	2.38E-01	3.11E-01	3.68E-01	4.12E-01	4.46E-01
942360	0.0	4.59E-09	2.39E-08	6.51E-08	1.34E-07	2.33E-07
942380	0.0	7.32E-05	3.87E-04	1.07E-03	2.24E-03	3.97E-03
942390	0.0	7.54E 00	1.72E 01	2.85E 01	4.08E 01	5.38E 01
942400	0.0	9.44E-03	4.99E-02	1.38E-01	2.88E-01	5.08E-01
942410	0.0	9.75E-06	1.18E-04	5.43E-04	1.63E-03	3.81E-03
942420	0.0	5.37E-09	1.51E-07	1.16E-06	5.08E-06	1.60E-05
942430	0.0	3.23E-14	1.19E-12	1.09E-11	5.34E-11	1.83E-10
952410	0.0	3.85E-08	8.32E-07	5.47E-06	2.14E-05	6.20E-05
952421	0.0	7.44E-12	3.80E-10	4.25E-09	2.43E-08	9.43E-08
952420	0.0	1.71E-12	4.85E-11	3.79E-10	1.66E-09	5.25E-09
952430	0.0	2.34E-12	1.52E-10	1.96E-09	1.25E-08	5.28E-08
952440	0.0	1.88E-18	1.60E-16	2.46E-15	1.76E-14	8.10E-14
962420	0.0	2.93E-11	1.42E-09	1.50E-08	8.18E-08	3.03E-07
962430	0.0	1.13E-14	1.31E-12	2.38E-11	1.89E-10	9.42E-10
962440	0.0	1.30E-15	1.96E-13	4.25E-12	3.92E-11	2.22E-10
FLUX		6.46E 14	8.46E 14	1.00E 15	1.13E 15	1.23E 15

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U235	2.87E 03	2.82E 03	2.82E 03	2.82E 03	2.83E 03	2.83E 03	2.84E 03
U236	0.0	3.73E 01	3.73E 01	3.73E 01	3.73E 01	3.73E 01	3.77E 01
U238	9.97E 05	9.82E 05	9.81E 05	9.81E 05	9.81E 05	9.81E 05	9.81E 05
NP237	0.0	4.91E 01	5.02E 01	5.02E 01	5.02E 01	5.02E 01	5.02E 01
PU239	0.0	1.29E 04	1.30E 04	1.30E 04	1.30E 04	1.30E 04	1.29E 04
PU240	0.0	1.22E 02	1.22E 02	1.22E 02	1.22E 02	1.22E 02	1.22E 02
SUBTOT	1.00E 06	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05
TOTALS	1.00E 06	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
HE	0.0	4.21E-03	4.74E-03	5.81E-03	6.88E-03	2.37E-02	1.99E-01
TL	0.0	1.60E-14	1.94E-14	2.98E-14	4.26E-14	4.17E-13	1.23E-12
PB	0.0	1.46E-09	1.94E-09	3.25E-09	5.19E-09	1.88E-07	1.11E-05
BI	0.0	8.84E-13	1.07E-12	1.65E-12	2.36E-12	2.31E-11	6.80E-11
PO	0.0	3.72E-17	4.53E-17	6.93E-17	9.92E-17	9.70E-16	2.86E-15
RN	0.0	1.41E-14	1.72E-14	2.63E-14	3.77E-14	3.69E-13	1.09E-12
RA	0.0	8.08E-11	9.84E-11	1.51E-10	2.16E-10	2.11E-09	6.21E-09
TH	0.0	9.67E-07	1.09E-06	1.34E-06	1.60E-06	5.84E-06	5.46E-05
PA	0.0	4.20E-08	4.36E-08	4.77E-08	5.21E-08	1.22E-07	8.55E-07
U	1.00E 06	9.84E 05	9.84E 05	9.84E 05	9.84E 05	9.84E 05	9.84E 05
NP	0.0	1.56E 02	5.02E 01	5.02E 01	5.02E 01	5.02E 01	5.02E 01
PU	0.0	1.30E 04	1.31E 04	1.31E 04	1.31E 04	1.31E 04	1.31E 04
AM	0.0	1.50E-02	1.90E-02	2.70E-02	3.49E-02	1.50E-01	7.27E-01
CM	0.0	7.36E-05	6.57E-05	5.10E-05	3.96E-05	1.02E-06	1.85E-07
TOTALS	1.00E 06	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05	9.97E 05

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U236	0.0	2.37E-03	2.37E-03	2.37E-03	2.37E-03	2.37E-03	2.39E-03
U237	0.0	9.38E 04	4.31E 03	9.08E 00	1.92E-02	0.0	0.0
U238	3.32E-01	3.27E-01	3.27E-01	3.27E-01	3.27E-01	3.27E-01	3.27E-01
NP237	0.0	3.46E-02	3.54E-02	3.54E-02	3.54E-02	3.54E-02	3.54E-02
NP238	0.0	3.16E 03	1.58E-01	3.97E-10	0.0	0.0	0.0
NP239	0.0	2.48E 07	3.58E 03	2.48E-06	2.48E-06	2.48E-06	2.47E-06
PU236	0.0	2.92E-02	2.88E-02	2.77E-02	2.66E-02	1.42E-02	1.99E-05
PU238	0.0	1.59E 01	1.61E 01	1.61E 01	1.61E 01	1.58E 01	1.28E 01
PU239	0.0	7.88E 02	7.95E 02	7.95E 02	7.95E 02	7.95E 02	7.94E 02
PU240	0.0	2.69E 01	2.69E 01	2.69E 01	2.69E 01	2.69E 01	2.68E 01
PU241	0.0	1.05E 02	1.04E 02	1.03E 02	1.03E 02	8.93E 01	2.12E 01
AM241	0.0	4.84E-02	6.14E-02	8.72E-02	1.13E-01	4.87E-01	2.35E 00
CM242	0.0	2.43E-01	2.17E-01	1.68E-01	1.30E-01	2.51E-03	1.59E-04
SUBTOT	3.32E-01	2.49E 07	8.83E 03	9.51E 02	9.41E 02	9.28E 02	8.58E 02
TOTALS	3.32E-01	4.97E 07	8.83E 03	9.51E 02	9.41E 02	9.28E 02	8.58E 02

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	2.01E 02	9.24E 00	1.95E-02	4.11E-05	0.0	0.0
NP239	0.0	7.35E 04	1.06E 01	7.34E-09	7.34E-09	7.34E-09	7.32E-09
PU238	0.0	5.28E-01	5.34E-01	5.34E-01	5.33E-01	5.22E-01	4.23E-01
PU239	0.0	2.45E 01	2.47E 01	2.47E 01	2.47E 01	2.47E 01	2.47E 01
PU240	0.0	8.38E-01	8.38E-01	8.38E-01	8.38E-01	8.37E-01	8.35E-01
SUBTOT	0.0	7.37E 04	4.59E 01	2.61E 01	2.61E 01	2.61E 01	2.59E 01
TOTALS	8.41E-03	1.33E 05	4.60E 01	2.61E 01	2.61E 01	2.61E 01	2.60E 01

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	1.43E 02	6.56E 00	1.38E-02	2.92E-05	0.0	0.0
NP239	0.0	4.78E 04	6.90E 00	4.77E-09	4.77E-09	4.77E-09	4.76E-09
SUBTOT	0.0	4.79E 04	1.35E 01	1.38E-02	2.92E-05	4.77E-09	4.76E-09
TOTALS	0.0	4.79E 04	1.35E 01	1.43E-02	3.65E-04	9.04E-06	7.88E-06

AI REFERENCE OXIDE LMFBR AXIAL BLANKET
POWER= 4.63 MW/MT, BURNUP= 2500. MWD/MT, FLUX= 9.71E 14 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TL	0.0	9.12E-08	1.10E-07	1.70E-07	2.43E-07	2.38E-06	7.01E-06
PB	0.0	2.29E-08	2.78E-08	4.28E-08	6.12E-08	5.99E-07	1.76E-06
BI	0.0	2.08E-07	2.52E-07	3.88E-07	5.55E-07	5.43E-06	1.60E-05
PO	0.0	9.68E-07	1.18E-06	1.81E-06	2.58E-06	2.53E-05	7.45E-05
RN	0.0	4.91E-07	5.97E-07	9.15E-07	1.31E-06	1.28E-05	3.77E-05
RA	0.0	4.44E-07	5.40E-07	8.27E-07	1.18E-06	1.16E-05	3.41E-05
TH	0.0	5.86E-07	6.21E-07	9.01E-07	1.25E-06	1.11E-05	3.27E-05
PA	0.0	6.71E-08	2.68E-08	5.93E-09	1.36E-09	1.78E-10	1.24E-09
U	8.41E-03	5.90E 04	9.25E 00	2.79E-02	8.41E-03	8.39E-03	8.43E-03
NP	0.0	7.35E 04	1.06E 01	7.34E-09	7.34E-09	7.34E-09	7.32E-09
PU	0.0	2.59E 01	2.61E 01	2.61E 01	2.61E 01	2.61E 01	2.59E 01
AM	0.0	1.75E-03	2.05E-03	2.91E-03	3.77E-03	1.63E-02	7.86E-02
CM	0.0	8.94E-04	7.98E-04	6.18E-04	4.79E-04	9.28E-06	6.09E-07
TOTALS	8.41E-03	1.33E 05	4.60E 01	2.61E 01	2.61E 01	2.61E 01	2.60E 01

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

CONCENTRATION HISTORY OF ACTINIDES DURING REACTOR LIFE

NUCLIDE CONCENTRATIONS, GRAM ATOMS / METRIC TON (U + PU)

	CHARGE	194. D	388. D	582. D	776. D	970. D
20040	0.0	1.17E-04	5.36E-04	1.37E-03	2.68E-03	4.53E-03
812080	0.0	1.11E-18	2.07E-17	1.25E-16	4.51E-16	1.20E-15
822080	0.0	3.94E-14	1.35E-12	1.21E-11	5.78E-11	1.94E-10
822120	0.0	6.37E-16	1.18E-14	7.18E-14	2.58E-13	6.86E-13
832120	0.0	6.05E-17	1.12E-15	6.81E-15	2.45E-14	6.51E-14
842120	0.0	1.06E-24	1.98E-23	1.20E-22	4.31E-22	1.15E-21
842160	0.0	2.49E-21	4.63E-20	2.81E-19	1.01E-18	2.68E-18
862200	0.0	9.31E-19	1.73E-17	1.05E-16	3.77E-16	1.00E-15
882240	0.0	5.23E-15	9.71E-14	5.89E-13	2.12E-12	5.63E-12
902280	0.0	1.00E-12	1.86E-11	1.13E-10	4.06E-10	1.08E-09
902290	0.0	4.05E-13	1.83E-12	4.86E-12	1.09E-11	2.32E-11
902300	0.0	8.63E-11	3.94E-10	1.00E-09	1.97E-09	3.35E-09
902310	0.0	3.52E-13	3.71E-13	4.21E-13	5.13E-13	6.54E-13
902320	0.0	2.37E-10	1.08E-09	2.71E-09	5.24E-09	8.71E-09
902330	0.0	1.47E-16	1.07E-15	3.33E-15	7.26E-15	1.31E-14
912310	0.0	4.38E-11	8.75E-11	1.34E-10	1.88E-10	2.55E-10
912320	0.0	4.21E-15	1.34E-14	2.55E-14	4.04E-14	5.90E-14
912330	0.0	1.77E-13	1.52E-12	5.07E-12	1.15E-11	2.11E-11
912341	0.0	5.35E-21	7.35E-20	3.04E-19	7.76E-19	1.55E-18
912340	0.0	1.77E-18	2.43E-17	1.00E-16	2.56E-16	5.10E-16
922320	0.0	8.08E-10	8.79E-09	3.76E-08	1.05E-07	2.29E-07
922330	0.0	3.58E-07	8.99E-07	1.52E-06	2.15E-06	2.77E-06
922340	0.0	1.16E-04	3.02E-04	5.35E-04	8.11E-04	1.13E-03
922350	8.51E 00	8.48E 00	8.42E 00	8.36E 00	8.28E 00	8.20E 00
922360	0.0	3.07E-02	7.87E-02	1.36E-01	1.99E-01	2.63E-01
922370	0.0	2.83E-03	4.60E-03	5.68E-03	6.39E-03	6.89E-03
922380	4.20E 03	4.18E 03	4.15E 03	4.12E 03	4.08E 03	4.04E 03
922390	0.0	1.87E-03	2.97E-03	3.64E-03	4.07E-03	4.36E-03
932360	0.0	4.64E-10	1.91E-09	4.09E-09	6.75E-09	9.72E-09
932370	0.0	5.71E-02	1.47E-01	2.54E-01	3.72E-01	4.95E-01
932380	0.0	8.34E-06	3.42E-05	7.35E-05	1.21E-04	1.75E-04
932390	0.0	2.69E-01	4.28E-01	5.24E-01	5.86E-01	6.28E-01
942360	0.0	1.85E-08	1.19E-07	3.51E-07	7.35E-07	1.27E-06
942380	0.0	3.00E-04	1.99E-03	6.01E-03	1.29E-02	2.30E-02
942390	0.0	1.52E 01	3.86E 01	6.58E 01	9.44E 01	1.23E 02
942400	0.0	3.87E-02	2.55E-01	7.70E-01	1.65E 00	2.92E 00
942410	0.0	8.05E-05	1.36E-03	7.09E-03	2.21E-02	5.20E-02
942420	0.0	9.04E-08	3.99E-06	3.69E-05	1.73E-04	5.55E-04
942430	0.0	6.14E-13	4.34E-11	4.96E-10	2.62E-09	9.11E-09
952410	0.0	5.72E-07	1.59E-05	1.20E-04	4.98E-04	1.47E-03
952421	0.0	2.24E-10	1.67E-08	2.25E-07	1.39E-06	5.52E-06
952420	0.0	2.88E-11	1.28E-09	1.19E-08	5.59E-08	1.79E-07
952430	0.0	8.00E-11	9.23E-09	1.51E-07	1.06E-06	4.63E-06
952440	0.0	7.30E-17	1.35E-14	2.72E-13	2.15E-12	1.02E-11
962420	0.0	8.38E-10	5.77E-08	7.15E-07	4.08E-06	1.51E-05
962430	0.0	6.62E-13	1.25E-10	2.78E-09	2.38E-08	1.20E-07
962440	0.0	9.08E-14	2.72E-11	7.86E-10	8.19E-09	4.85E-08
FLUX		7.30E 14	1.17E 15	1.44E 15	1.63E 15	1.76E 15

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U235	2.00E 03	1.93E 03	1.93E 03	1.93E 03	1.93E 03	1.93E 03	1.95E 03
U236	0.0	6.21E 01	6.21E 01	6.21E 01	6.22E 01	6.23E 01	6.43E 01
U238	1.00E 06	9.61E 05	9.61E 05	9.61E 05	9.61E 05	9.61E 05	9.61E 05
NP237	0.0	1.17E 02	1.19E 02	1.19E 02	1.19E 02	1.19E 02	1.19E 02
PU238	0.0	5.48E 00	5.52E 00	5.52E 00	5.51E 00	5.40E 00	4.38E 00
PU239	0.0	2.94E 04	2.96E 04	2.96E 04	2.96E 04	2.96E 04	2.95E 04
PU240	0.0	7.02E 02	7.02E 02	7.02E 02	7.02E 02	7.01E 02	6.99E 02
PU241	0.0	1.25E 01	1.25E 01	1.24E 01	1.23E 01	1.07E 01	2.53E 00
SUBTOT	1.00E 06	9.93E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05
TOTALS	1.00E 06	9.94E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
HE	0.0	1.81E-02	1.95E-02	2.21E-02	2.47E-02	6.62E-02	5.00E-01
TL	0.0	2.49E-13	2.75E-13	3.60E-13	4.54E-13	2.72E-12	7.15E-12
PB	0.0	4.05E-08	4.76E-08	6.47E-08	8.65E-08	1.38E-06	6.57E-05
BI	0.0	1.38E-11	1.52E-11	1.99E-11	2.51E-11	1.51E-10	3.96E-10
PO	0.0	5.80E-16	6.49E-16	8.38E-16	1.06E-15	6.34E-15	1.66E-14
RN	0.0	2.20E-13	2.47E-13	3.19E-13	4.01E-13	2.41E-12	6.32E-12
RA	0.0	1.26E-09	1.41E-09	1.82E-09	2.29E-09	1.38E-08	3.62E-08
TH	0.0	3.04E-06	3.29E-06	3.78E-06	4.29E-06	1.35E-05	1.35E-04
PA	0.0	6.37E-08	6.27E-08	6.39E-08	6.66E-08	1.14E-07	6.17E-07
U	1.00E 06	9.63E 05	9.63E 05	9.63E 05	9.63E 05	9.63E 05	9.63E 05
NP	0.0	2.67E 02	1.19E 02	1.19E 02	1.19E 02	1.19E 02	1.19E 02
PU	0.0	3.01E 04	3.03E 04	3.03E 04	3.03E 04	3.03E 04	3.02E 04
AM	0.0	3.57E-01	4.12E-01	5.20E-01	6.28E-01	2.20E 00	1.01E 01
CM	0.0	3.70E-03	3.29E-03	2.56E-03	1.99E-03	7.60E-05	2.18E-05
TOTALS	1.00E 06	9.94E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05	9.94E 05

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	1.33E 05	6.13E 03	1.29E 01	2.73E-02	0.0	0.0
U238	3.33E-01	3.20E-01	3.20E-01	3.20E-01	3.20E-01	3.20E-01	3.20E-01
NP237	0.0	8.28E-02	8.39E-02	8.39E-02	8.39E-02	8.39E-02	8.41E-02
NP238	0.0	1.09E 04	5.45E-01	1.36E-09	0.0	0.0	0.0
NP239	0.0	3.49E 07	5.04E 03	2.17E-04	2.17E-04	2.17E-04	2.16E-04
PU236	0.0	1.60E-01	1.57E-01	1.51E-01	1.45E-01	7.73E-02	1.09E-04
PU238	0.0	9.26E 01	9.33E 01	9.32E 01	9.31E 01	9.12E 01	7.39E 01
PU239	0.0	1.80E 03	1.81E 03	1.81E 03	1.81E 03	1.81E 03	1.81E 03
PU240	0.0	1.55E 02	1.55E 02	1.55E 02	1.55E 02	1.55E 02	1.54E 02
PU241	0.0	1.43E 03	1.42E 03	1.41E 03	1.40E 03	1.22E 03	2.89E 02
AM241	0.0	1.15E 00	1.33E 00	1.68E 00	2.03E 00	7.13E 00	3.26E 01
AM242M	0.0	1.30E-02	1.30E-02	1.30E-02	1.30E-02	1.28E-02	1.13E-02
AM242	0.0	3.50E 01	1.30E-02	1.30E-02	1.30E-02	1.28E-02	1.13E-02
CM242	0.0	1.21E 01	1.08E 01	8.34E 00	6.46E 00	1.26E-01	9.29E-03
SUBTOT	3.33E-01	3.50E 07	1.47E 04	3.50E 03	3.47E 03	3.29E 03	2.36E 03
TOTALS	3.33E-01	7.00E 07	1.47E 04	3.50E 03	3.47E 03	3.29E 03	2.36E 03

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	2.86E 02	1.31E 01	2.77E-02	5.85E-05	0.0	0.0
NP239	0.0	1.03E 05	1.49E 01	6.43E-07	6.43E-07	6.42E-07	6.41E-07
PU238	0.0	3.07E 00	3.09E 00	3.09E 00	3.08E 00	3.02E 00	2.45E 00
PU239	0.0	5.61E 01	5.63E 01	5.63E 01	5.63E 01	5.63E 01	5.63E 01
PU240	0.0	4.82E 00	4.82E 00	4.82E 00	4.82E 00	4.82E 00	4.80E 00
SUBTOT	0.0	1.04E 05	9.23E 01	6.43E 01	6.42E 01	6.42E 01	6.35E 01
TOTALS	8.43E-03	1.87E 05	9.25E 01	6.44E 01	6.44E 01	6.45E 01	6.46E 01

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
U237	0.0	2.03E 02	9.34E 00	1.97E-02	4.15E-05	0.0	0.0
NP239	0.0	6.72E 04	9.71E 00	4.18E-07	4.18E-07	4.18E-07	4.17E-07
SUBTOT	0.0	6.74E 04	1.90E 01	1.97E-02	4.19E-05	4.18E-07	4.17E-07
TOTALS	0.0	6.75E 04	1.91E 01	4.12E-02	1.57E-02	3.44E-04	6.93E-05

AI REFERENCE OXIDE LMFBR RADIAL BLANKET
POWER= 8.35 MW/MT, BURNUP= 8100. MWD/MT, FLUX= 1.35E 15 N/CM**2-SEC

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
TL	0.0	1.42E-06	1.57E-06	2.06E-06	2.59E-06	1.55E-05	4.08E-05
PB	0.0	3.58E-07	3.94E-07	5.17E-07	6.51E-07	3.91E-06	1.03E-05
BI	0.0	3.24E-06	3.57E-06	4.69E-06	5.90E-06	3.55E-05	9.31E-05
PO	0.0	1.51E-05	1.68E-05	2.18E-05	2.75E-05	1.65E-04	4.33E-04
RN	0.0	7.66E-06	8.57E-06	1.11E-05	1.39E-05	8.37E-05	2.20E-04
RA	0.0	6.92E-06	7.75E-06	1.00E-05	1.26E-05	7.57E-05	1.99E-04
TH	0.0	7.01E-06	7.64E-06	9.82E-06	1.23E-05	7.24E-05	1.90E-04
PA	0.0	2.79E-07	1.20E-07	2.63E-08	5.85E-09	1.66E-10	8.97E-10
U	8.43E-03	8.31E 04	1.32E 01	3.60E-02	8.38E-03	8.42E-03	8.65E-03
NP	0.0	1.03E 05	1.49E 01	6.43E-07	6.43E-07	6.42E-07	6.41E-07
PU	0.0	6.40E 01	6.43E 01	6.43E 01	6.43E 01	6.42E 01	6.36E 01
AM	0.0	4.31E-02	4.43E-02	5.60E-02	6.77E-02	2.38E-01	1.09E 00
CM	0.0	4.46E-02	3.96E-02	3.07E-02	2.38E-02	4.72E-04	3.78E-05
TOTALS	8.43E-03	1.87E 05	9.25E 01	6.44E 01	6.44E 01	6.45E 01	6.46E 01

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U234	1.38E 01	1.46E 01	1.60E 01	1.75E 01	4.07E 01	2.58E 02
U235	1.04E 03	1.04E 03	1.04E 03	1.04E 03	1.05E 03	1.11E 03
U236	1.92E 01	1.95E 01	2.00E 01	2.06E 01	2.93E 01	1.19E 02
U238	8.15E 05	8.15E 05	8.15E 05	8.15E 05	8.15E 05	8.15E 05
NP237	1.29E 02	1.32E 02	1.32E 02	1.33E 02	1.39E 02	3.70E 02
PU238	1.15E 03	1.16E 03	1.16E 03	1.17E 03	1.16E 03	9.44E 02
PU239	7.82E 04	7.85E 04	7.85E 04	7.85E 04	7.85E 04	7.84E 04
PU240	3.32E 04	3.32E 04	3.32E 04	3.32E 04	3.32E 04	3.31E 04
PU241	9.20E 03	9.16E 03	9.08E 03	9.00E 03	7.84E 03	1.86E 03
PU242	5.70E 03	5.70E 03	5.70E 03	5.70E 03	5.70E 03	5.70E 03
AM241	8.04E 02	8.44E 02	9.24E 02	1.00E 03	2.16E 03	7.90E 03
AM242M	1.56E 01	1.56E 01	1.55E 01	1.55E 01	1.53E 01	1.36E 01
AM243	4.49E 02	4.49E 02	4.49E 02	4.49E 02	4.49E 02	4.48E 02
CM242	3.90E 01	3.45E 01	2.68E 01	2.07E 01	4.08E-01	3.27E-02
CM244	2.68E 01	2.67E 01	2.65E 01	2.64E 01	2.39E 01	8.49E 00
SUBTOT	9.45E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05
TOTALS	9.46E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
HE	1.05E 00	1.16E 00	1.33E 00	1.48E 00	2.61E 00	1.31E 01
TL	6.81E-13	7.72E-13	1.07E-12	1.41E-12	9.87E-12	2.68E-11
PB	8.41E-08	1.04E-07	1.54E-07	2.20E-07	4.79E-06	2.45E-04
BI	3.77E-11	4.27E-11	5.94E-11	7.81E-11	5.46E-10	1.48E-09
PO	1.59E-15	1.83E-15	2.50E-15	3.28E-15	2.30E-14	6.23E-14
RN	6.03E-13	6.96E-13	9.50E-13	1.25E-12	8.73E-12	2.37E-11
RA	3.45E-09	3.98E-09	5.43E-09	7.13E-09	4.99E-08	1.35E-07
TH	3.03E-05	3.36E-05	4.09E-05	4.90E-05	2.67E-04	1.17E-02
PA	9.85E-07	9.94E-07	9.95E-07	9.96E-07	1.02E-06	1.30E-06
U	8.16E 05	8.16E 05	8.16E 05	8.16E 05	8.16E 05	8.17E 05
NP	3.90E 02	1.32E 02	1.32E 02	1.33E 02	1.39E 02	3.70E 02
PU	1.27E 05	1.28E 05	1.28E 05	1.28E 05	1.26E 05	1.20E 05
AM	1.27E 03	1.31E 03	1.39E 03	1.47E 03	2.62E 03	8.37E 03
CM	6.72E 01	6.27E 01	5.47E 01	4.86E 01	2.57E 01	9.29E 00
TOTALS	9.46E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05	9.46E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U234	8.56E-02	9.01E-02	9.92E-02	1.08E-01	2.52E-01	1.59E 00
U237	2.25E 05	1.04E 04	2.18E 01	4.61E-02	0.0	0.0
U238	2.72E-01	2.72E-01	2.72E-01	2.72E-01	2.72E-01	2.72E-01
NP237	9.11E-02	9.31E-02	9.34E-02	9.35E-02	9.79E-02	2.61E-01
NP238	3.33E 04	1.67E 00	4.18E-09	1.05E-17	0.0	0.0
NP239	6.07E 07	8.85E 03	8.65E 01	8.65E 01	8.65E 01	8.62E 01
PU236	6.09E-01	5.99E-01	5.76E-01	5.53E-01	2.95E-01	4.15E-04
PU238	1.94E 04	1.95E 04	1.96E 04	1.97E 04	1.96E 04	1.59E 04
PU239	4.79E 03	4.81E 03	4.81E 03	4.81E 03	4.81E 03	4.81E 03
PU240	7.32E 03	7.32E 03	7.32E 03	7.32E 03	7.31E 03	7.30E 03
PU241	1.05E 06	1.05E 06	1.04E 06	1.03E 06	8.95E 05	2.12E 05
PU242	2.22E 01	2.22E 01	2.22E 01	2.22E 01	2.22E 01	2.22E 01
AM241	2.60E 03	2.73E 03	2.99E 03	3.25E 03	6.99E 03	2.56E 04
AM242M	1.51E 02	1.51E 02	1.51E 02	1.51E 02	1.49E 02	1.32E 02
AM242	2.49E 05	1.51E 02	1.51E 02	1.51E 02	1.49E 02	1.32E 02
AM243	8.64E 01	8.65E 01	8.65E 01	8.65E 01	8.65E 01	8.62E 01
CM242	1.29E 05	1.14E 05	8.86E 04	6.87E 04	1.35E 03	1.08E 02
CM243	6.75E 01	6.74E 01	6.72E 01	6.69E 01	6.33E 01	3.53E 01
CM244	2.17E 03	2.16E 03	2.15E 03	2.13E 03	1.93E 03	6.87E 02
SUBTOT	6.24E 07	1.22E 06	1.16E 06	1.13E 06	9.38E 05	2.67E 05
TOTALS	1.24E 08	1.22E 06	1.16E 06	1.13E 06	9.38E 05	2.67E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U237	4.84E 02	2.22E 01	4.69E-02	9.88E-05	0.0	0.0
NP239	1.80E 05	2.62E 01	2.56E-01	2.56E-01	2.56E-01	2.56E-01
PU236	2.12E-02	2.08E-02	2.00E-02	1.92E-02	1.03E-02	1.44E-05
PU238	6.44E 02	6.46E 02	6.50E 02	6.52E 02	6.50E 02	5.28E 02
PU239	1.49E 02	1.50E 02	1.50E 02	1.50E 02	1.50E 02	1.49E 02
PU240	2.28E 02	2.28E 02	2.28E 02	2.28E 02	2.28E 02	2.27E 02
PU241	4.36E 01	4.34E 01	4.30E 01	4.26E 01	3.71E 01	8.80E 00
PU242	6.57E-01	6.57E-01	6.57E-01	6.57E-01	6.57E-01	6.57E-01
AM241	8.69E 01	9.13E 01	9.99E 01	1.08E 02	2.33E 02	8.55E 02
AM242M	4.30E-02	4.30E-02	4.30E-02	4.30E-02	4.25E-02	3.75E-02
AM243	2.78E-01	2.78E-01	2.78E-01	2.78E-01	2.78E-01	2.78E-01
CM242	4.75E 02	4.21E 02	3.26E 02	2.53E 02	4.97E 00	3.98E-01
CM243	2.47E-01	2.46E-01	2.45E-01	2.44E-01	2.31E-01	1.29E-01
CM244	7.58E 00	7.56E 00	7.51E 00	7.47E 00	6.76E 00	2.40E 00
SUBTOT	1.82E 05	1.64E 03	1.51E 03	1.44E 03	1.31E 03	1.77E 03
TOTALS	3.28E 05	1.64E 03	1.51E 03	1.44E 03	1.31E 03	1.77E 03

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET
POWER= 95.41 MW/MT, BURNUP= 51519. MWD/MT, FLUX= 3.62E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U237	3.43E 02	1.58E 01	3.33E-02	7.02E-05	0.0	0.0
NP239	1.17E 05	1.71E 01	1.67E-01	1.67E-01	1.67E-01	1.66E-01
AM243	2.50E-01	2.51E-01	2.51E-01	2.50E-01	2.50E-01	2.50E-01
CM242	3.33E 02	2.95E 02	2.28E 02	1.77E 02	3.48E 00	2.79E-01
CM243	2.47E-02	2.46E-02	2.45E-02	2.44E-02	2.31E-02	1.29E-02
CM244	1.52E 00	1.51E 00	1.50E 00	1.49E 00	1.35E 00	4.81E-01
SUBTOT	1.18E 05	3.29E 02	2.30E 02	1.79E 02	5.27E 00	1.19E 00
TOTALS	1.18E 05	3.29E 02	2.30E 02	1.79E 02	5.28E 00	1.20E 00

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL BLANKET

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
TL	3.89E-06	4.40E-06	6.13E-06	8.05E-06	5.63E-05	1.53E-04
PB	9.79E-07	1.11E-06	1.54E-06	2.03E-06	1.42E-05	3.85E-05
BI	8.87E-06	1.00E-05	1.40E-05	1.84E-05	1.28E-04	3.49E-04
PO	4.13E-05	4.73E-05	6.51E-05	8.55E-05	5.98E-04	1.62E-03
RN	2.09E-05	2.42E-05	3.30E-05	4.33E-05	3.03E-04	8.23E-04
RA	1.89E-05	2.18E-05	2.98E-05	3.92E-05	2.74E-04	7.44E-04
TH	2.45E-05	2.15E-05	2.93E-05	3.83E-05	2.62E-04	7.17E-04
PA	1.07E-06	3.16E-08	8.06E-09	2.90E-09	1.49E-09	1.89E-09
U	1.45E 05	2.22E 01	5.68E-02	1.03E-02	1.47E-02	5.37E-02
NP	1.80E 05	2.62E 01	2.56E-01	2.56E-01	2.56E-01	2.56E-01
PU	2.15E 03	1.07E 03	1.07E 03	1.07E 03	1.07E 03	9.14E 02
AM	4.12E 02	9.16E 01	1.00E 02	1.09E 02	2.34E 02	8.55E 02
CM	4.83E 02	4.29E 02	3.34E 02	2.61E 02	1.20E 01	2.93E 00
TOTALS	3.28E 05	1.64E 03	1.51E 03	1.44E 03	1.31E 03	1.77E 03

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U234	8.04E 00	8.46E 00	9.30E 00	1.01E 01	2.35E 01	1.48E 02
U235	1.42E 03	1.42E 03	1.42E 03	1.42E 03	1.43E 03	1.47E 03
U236	3.75E 01	3.77E 01	3.80E 01	3.83E 01	4.34E 01	9.59E 01
U238	8.78E 05	8.78E 05	8.78E 05	8.78E 05	8.78E 05	8.78E 05
NP237	1.24E 02	1.26E 02	1.27E 02	1.27E 02	1.30E 02	2.63E 02
PU238	6.62E 02	6.65E 02	6.68E 02	6.71E 02	6.59E 02	5.43E 02
PU239	5.73E 04	5.76E 04	5.76E 04	5.76E 04	5.76E 04	5.75E 04
PU240	1.93E 04	1.93E 04	1.93E 04	1.93E 04	1.93E 04	1.93E 04
PU241	5.28E 03	5.25E 03	5.21E 03	5.16E 03	4.50E 03	1.07E 03
PU242	3.27E 03	3.27E 03	3.27E 03	3.27E 03	3.27E 03	3.27E 03
AM241	4.61E 02	4.84E 02	5.29E 02	5.75E 02	1.24E 03	4.53E 03
AM242M	8.92E 00	8.91E 00	8.91E 00	8.90E 00	8.79E 00	7.78E 00
AM243	2.57E 02	2.58E 02	2.58E 02	2.58E 02	2.57E 02	2.57E 02
CM242	2.23E 01	1.98E 01	1.53E 01	1.19E 01	2.34E-01	1.87E-02
CM244	1.53E 01	1.53E 01	1.52E 01	1.51E 01	1.37E 01	4.86E 00
SUBTOT	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05
TOTALS	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
HE	6.12E-01	6.71E-01	7.72E-01	8.58E-01	1.52E 00	7.69E 00
TL	4.97E-13	5.68E-13	7.69E-13	1.00E-12	6.82E-12	1.84E-11
PB	6.54E-08	7.98E-08	1.16E-07	1.63E-07	3.34E-06	1.68E-04
BI	2.75E-11	3.14E-11	4.26E-11	5.55E-11	3.77E-10	1.02E-09
PO	1.16E-15	1.33E-15	1.79E-15	2.33E-15	1.59E-14	4.28E-14
RN	4.39E-13	5.04E-13	6.80E-13	8.86E-13	6.03E-12	1.63E-11
RA	2.51E-09	2.88E-09	3.89E-09	5.07E-09	3.45E-08	9.31E-08
TH	1.86E-05	2.07E-05	2.51E-05	2.99E-05	1.59E-04	6.78E-03
PA	5.92E-07	5.96E-07	5.97E-07	5.99E-07	6.34E-07	1.01E-06
U	8.79E 05	8.79E 05	8.79E 05	8.79E 05	8.79E 05	8.79E 05
NP	3.38E 02	1.26E 02	1.27E 02	1.27E 02	1.30E 02	2.63E 02
PU	8.59E 04	8.61E 04	8.60E 04	8.60E 04	8.53E 04	8.17E 04
AM	7.27E 02	7.50E 02	7.96E 02	8.41E 02	1.50E 03	4.80E 03
CM	3.85E 01	3.59E 01	3.14E 01	2.78E 01	1.47E 01	5.32E 00
TOTALS	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05	9.66E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U234	4.98E-02	5.23E-02	5.75E-02	6.27E-02	1.45E-01	9.17E-01
U237	1.86E 05	8.55E 03	1.80E 01	3.80E-02	0.0	0.0
U238	2.92E-01	2.92E-01	2.92E-01	2.92E-01	2.92E-01	2.92E-01
NP237	8.76E-02	8.92E-02	8.93E-02	8.94E-02	9.19E-02	1.85E-01
NP238	2.37E 04	1.19E 00	2.98E-09	7.46E-18	0.0	0.0
NP239	4.97E 07	7.22E 03	4.95E 01	4.95E 01	4.95E 01	4.94E 01
PU236	4.17E-01	4.10E-01	3.94E-01	3.79E-01	2.02E-01	2.84E-04
PU238	1.12E 04	1.12E 04	1.13E 04	1.13E 04	1.13E 04	9.16E 03
PU239	3.52E 03	3.53E 03	3.53E 03	3.53E 03	3.53E 03	3.53E 03
PU240	4.26E 03	4.26E 03	4.26E 03	4.26E 03	4.26E 03	4.25E 03
PU241	6.02E 05	6.00E 05	5.95E 05	5.89E 05	5.13E 05	1.22E 05
PU242	1.27E 01	1.27E 01	1.27E 01	1.27E 01	1.27E 01	1.27E 01
AM241	1.49E 03	1.57E 03	1.72E 03	1.86E 03	4.01E 03	1.47E 04
AM242M	8.67E 01	8.67E 01	8.66E 01	8.65E 01	8.55E 01	7.56E 01
AM242	1.42E 05	8.67E 01	8.66E 01	8.65E 01	8.55E 01	7.56E 01
AM243	4.95E 01	4.95E 01	4.95E 01	4.95E 01	4.95E 01	4.94E 01
CM242	7.39E 04	6.55E 04	5.08E 04	3.94E 04	7.74E 02	6.20E 01
CM243	3.87E 01	3.86E 01	3.85E 01	3.83E 01	3.63E 01	2.02E 01
CM244	1.24E 03	1.24E 03	1.23E 03	1.22E 03	1.11E 03	3.94E 02
SUBTOT	5.07E 07	7.03E 05	6.68E 05	6.51E 05	5.39E 05	1.54E 05
TOTALS	1.01E 08	7.03E 05	6.68E 05	6.51E 05	5.39E 05	1.54E 05

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE THERMAL POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U237	3.99E 02	1.83E 01	3.87E-02	8.16E-05	0.0	0.0
NP239	1.47E 05	2.14E 01	1.47E-01	1.47E-01	1.47E-01	1.46E-01
PU238	3.70E 02	3.72E 02	3.74E 02	3.75E 02	3.74E 02	3.03E 02
PU239	1.09E 02	1.10E 02	1.10E 02	1.10E 02	1.10E 02	1.10E 02
PU240	1.33E 02	1.33E 02	1.33E 02	1.33E 02	1.33E 02	1.32E 02
PU241	2.50E 01	2.49E 01	2.47E 01	2.45E 01	2.13E 01	5.05E 00
PU242	3.76E-01	3.76E-01	3.76E-01	3.76E-01	3.76E-01	3.76E-01
AM241	4.98E 01	5.23E 01	5.72E 01	6.21E 01	1.34E 02	4.90E 02
AM242M	2.47E-02	2.47E-02	2.46E-02	2.46E-02	2.43E-02	2.15E-02
AM243	1.59E-01	1.59E-01	1.59E-01	1.59E-01	1.59E-01	1.59E-01
CM242	2.72E 02	2.41E 02	1.87E 02	1.45E 02	2.85E 00	2.28E-01
CM243	1.41E-01	1.41E-01	1.41E-01	1.40E-01	1.32E-01	7.38E-02
CM244	4.34E 00	4.33E 00	4.30E 00	4.28E 00	3.87E 00	1.38E 00
SUBTOT	1.49E 05	9.77E 02	8.90E 02	8.54E 02	7.79E 02	1.04E 03
TOTALS	2.67E 05	9.77E 02	8.90E 02	8.54E 02	7.79E 02	1.04E 03

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS
POWER= 58.23 MW/MT, BURNUP= 32977. MWD/MT, FLUX= 2.65E 15 N/CM**2-SEC

NUCLIDE GAMMA POWER, WATTS / METRIC TON (U + PU)

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
U237	2.84E 02	1.30E 01	2.75E-02	5.79E-05	0.0	0.0
NP239	9.57E 04	1.39E 01	9.55E-02	9.54E-02	9.54E-02	9.52E-02
AM243	1.43E-01	1.44E-01	1.44E-01	1.44E-01	1.43E-01	1.43E-01
CM242	1.91E 02	1.69E 02	1.31E 02	1.01E 02	1.99E 00	1.60E-01
CM244	8.69E-01	8.66E-01	8.61E-01	8.56E-01	7.75E-01	2.75E-01
SUBTOT	9.61E 04	1.97E 02	1.32E 02	1.03E 02	3.01E 00	6.74E-01
TOTALS	9.62E 04	1.97E 02	1.32E 02	1.03E 02	3.03E 00	6.86E-01

AI REFERENCE OXIDE LMFBR-CORE PLUS AXIAL AND RADIAL BLANKETS

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	INITIAL	30. D	90. D	150. D	1096. D	10958. D
TL	2.84E-06	3.24E-06	4.39E-06	5.72E-06	3.89E-05	1.05E-04
PB	7.14E-07	8.16E-07	1.10E-06	1.44E-06	9.79E-06	2.64E-05
BI	6.47E-06	7.39E-06	1.00E-05	1.30E-05	8.88E-05	2.40E-04
PO	3.01E-05	3.45E-05	4.66E-05	6.07E-05	4.13E-04	1.12E-03
RN	1.53E-05	1.75E-05	2.36E-05	3.08E-05	2.09E-04	5.65E-04
RA	1.38E-05	1.58E-05	2.14E-05	2.78E-05	1.89E-04	5.11E-04
TH	1.70E-05	1.56E-05	2.10E-05	2.72E-05	1.81E-04	4.92E-04
PA	7.31E-07	6.92E-08	1.59E-08	4.16E-09	9.22E-10	1.47E-09
U	1.18E 05	1.84E 01	4.79E-02	9.49E-03	1.20E-02	3.44E-02
NP	1.47E 05	2.14E 01	1.47E-01	1.47E-01	1.47E-01	1.46E-01
PU	1.26E 03	6.39E 02	6.41E 02	6.42E 02	6.38E 02	5.51E 02
AM	2.36E 02	5.25E 01	5.74E 01	6.23E 01	1.34E 02	4.90E 02
CM	2.77E 02	2.46E 02	1.91E 02	1.49E 02	6.85E 00	1.68E 00
TOTALS	2.67E 05	9.77E 02	8.90E 02	8.54E 02	7.79E 02	1.04E 03

APPENDIX C: CONCENTRATIONS, RADIOACTIVITY, AND THERMAL POWER OF NUCLIDES PRODUCED FROM ACTIVATION OF THE CLADDING FOR THE CORE OF THE ATOMICS INTERNATIONAL REFERENCE OXIDE LMFBR

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

NUCLIDE CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30.0 D	90.0 D	150. D	1096. D	10958. D
HE 4	0.0	5.82E 00	5.82E 00	5.82E 00	5.82E 00	5.82E 00	5.82E 00
C 12	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02	2.36E 02
C 13	3.59E 00	2.09E 01	2.09E 01	2.09E 01	2.09E 01	2.09E 01	2.09E 01
O 16	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05
O 17	5.69E 01	7.74E 01	7.74E 01	7.74E 01	7.74E 01	7.74E 01	7.74E 01
O 18	3.32E 02	3.32E 02	3.32E 02	3.32E 02	3.32E 02	3.32E 02	3.32E 02
F 19	0.0	5.05E-02	5.05E-02	5.05E-02	5.05E-02	5.05E-02	5.05E-02
NA 23	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03
MG 24	0.0	9.66E-01	9.68E-01	9.68E-01	9.68E-01	9.68E-01	9.68E-01
MG 26	0.0	9.83E-02	9.83E-02	9.83E-02	9.83E-02	9.83E-02	9.83E-02
SI 28	2.06E 03	2.06E 03	2.06E 03	2.06E 03	2.06E 03	2.06E 03	2.06E 03
SI 29	1.09E 02	1.09E 02	1.09E 02	1.09E 02	1.09E 02	1.09E 02	1.09E 02
SI 30	7.41E 01	7.40E 01	7.40E 01	7.40E 01	7.40E 01	7.40E 01	7.40E 01
P 31	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02
S 32	8.51E 01	8.50E 01	8.50E 01	8.50E 01	8.50E 01	8.50E 01	8.50E 01
S 33	6.90E-01	6.92E-01	6.92E-01	6.92E-01	6.92E-01	6.92E-01	6.92E-01
S 34	4.01E 00	4.01E 00	4.01E 00	4.01E 00	4.01E 00	4.01E 00	4.01E 00
TI 47	0.0	5.27E-02	5.27E-02	5.27E-02	5.27E-02	5.27E-02	5.27E-02
TI 49	0.0	8.18E-02	8.18E-02	8.18E-02	8.18E-02	8.18E-02	8.18E-02
TI 50	0.0	3.84E-01	3.84E-01	3.84E-01	3.84E-01	3.84E-01	3.84E-01
V 50	0.0	1.52E 00	1.52E 00	1.52E 00	1.52E 00	1.52E 00	1.52E 00
V 51	0.0	7.79E 00	8.14E 00	8.39E 00	8.44E 00	8.46E 00	8.46E 00
CR 50	2.23E 03	2.22E 03	2.22E 03	2.22E 03	2.22E 03	2.22E 03	2.22E 03
CR 51	0.0	6.70E-01	3.17E-01	7.10E-02	1.59E-02	9.08E-13	0.0
CR 52	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04
CR 53	5.23E 03	5.05E 03	5.05E 03	5.05E 03	5.05E 03	5.05E 03	5.05E 03
CR 54	1.33E 03	1.55E 03	1.55E 03	1.55E 03	1.55E 03	1.57E 03	1.57E 03
MN 54	0.0	1.88E 01	1.76E 01	1.53E 01	1.34E 01	1.53E 00	2.46E-10
MN 55	5.98E 03	5.96E 03	5.96E 03	5.96E 03	5.96E 03	5.97E 03	5.99E 03
FE 54	1.19E 04	1.19E 04	1.19E 04	1.19E 04	1.19E 04	1.19E 04	1.19E 04
FE 55	0.0	3.04E 01	2.97E 01	2.85E 01	2.72E 01	1.37E 01	1.02E-02
FE 56	1.95E 05	1.94E 05	1.94E 05	1.94E 05	1.94E 05	1.94E 05	1.94E 05
FE 57	4.72E 03	5.17E 03	5.17E 03	5.17E 03	5.17E 03	5.17E 03	5.17E 03
FE 58	7.22E 02	8.11E 02	8.15E 02	8.19E 02	8.21E 02	8.24E 02	8.24E 02
FE 59	0.0	2.10E-01	1.32E-01	5.26E-02	2.09E-02	9.79E-09	0.0
CO 58	0.0	1.30E 01	9.71E 00	5.42E 00	3.02E 00	3.17E-04	0.0
CO 59	2.99E 02	3.00E 02	3.00E 02	3.00E 02	3.00E 02	3.00E 02	3.00E 02
CO 60	0.0	1.19E 00	1.18E 00	1.15E 00	1.13E 00	8.01E-01	2.28E-02
NI 58	1.60E 04	1.59E 04	1.59E 04	1.59E 04	1.59E 04	1.59E 04	1.59E 04
NI 59	0.0	2.81E 01	2.81E 01	2.81E 01	2.81E 01	2.81E 01	2.81E 01
NI 60	6.40E 03	6.39E 03	6.39E 03	6.39E 03	6.39E 03	6.39E 03	6.39E 03
NI 61	2.93E 02	3.02E 02	3.02E 02	3.02E 02	3.02E 02	3.02E 02	3.02E 02
NI 62	9.23E 02	9.23E 02	9.23E 02	9.23E 02	9.23E 02	9.23E 02	9.23E 02
NI 63	0.0	1.10E 00	1.10E 00	1.10E 00	1.09E 00	1.07E 00	8.76E-01
NI 64	2.84E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02	2.83E 02
CU 65	0.0	2.76E-01	2.76E-01	2.76E-01	2.76E-01	2.76E-01	2.76E-01
SUBTOT	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05
TOTALS	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

ELEMENT CONCENTRATIONS, GRAMS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30.0 D	90.0 D	150. D	1096. D	10958. D
H	0.0	3.39E-09	3.38E-09	3.35E-09	3.32E-09	2.86E-09	6.26E-10
HE	0.0	5.82E 00	5.82E 00	5.82E 00	5.82E 00	5.82E 00	5.82E 00
LI	0.0	1.38E-06	1.38E-06	1.38E-06	1.38E-06	1.38E-06	1.38E-06
BE	0.0	2.74E-03	2.74E-03	2.74E-03	2.74E-03	2.74E-03	2.74E-03
B	0.0	5.01E-07	5.01E-07	5.01E-07	5.01E-07	5.02E-07	5.06E-07
C	2.40E 02	2.57E 02	2.57E 02	2.57E 02	2.57E 02	2.57E 02	2.57E 02
N	0.0	8.75E-07	9.91E-07	1.25E-06	1.51E-06	5.61E-06	4.83E-05
O	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05	1.34E 05
F	0.0	5.05E-02	5.05E-02	5.05E-02	5.05E-02	5.05E-02	5.05E-02
NE	0.0	3.92E-02	3.92E-02	3.92E-02	3.92E-02	3.92E-02	3.92E-02
NA	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03	2.27E 03
MG	0.0	1.11E 00	1.11E 00	1.11E 00	1.11E 00	1.11E 00	1.11E 00
AL	0.0	2.78E-04	2.77E-04	2.77E-04	2.77E-04	2.77E-04	2.77E-04
SI	2.24E 03	2.24E 03	2.24E 03	2.24E 03	2.24E 03	2.24E 03	2.24E 03
P	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02
S	8.98E 01	8.97E 01	8.97E 01	8.97E 01	8.97E 01	8.97E 01	8.97E 01
CL	0.0	2.95E-08	3.15E-08	3.44E-08	3.61E-08	3.90E-08	3.90E-08
AR	0.0	1.75E-15	2.69E-16	2.69E-16	2.69E-16	2.69E-16	2.69E-16
K	0.0	2.27E-12	2.27E-12	2.27E-12	2.27E-12	2.27E-12	2.27E-12
CA	0.0	4.40E-06	4.40E-06	4.40E-06	4.40E-06	4.40E-06	4.40E-06
SC	0.0	1.14E-06	2.66E-09	1.47E-10	1.66E-10	2.38E-10	2.39E-10
TI	0.0	5.19E-01	5.19E-01	5.19E-01	5.19E-01	5.19E-01	5.19E-01
V	0.0	9.31E 00	9.66E 00	9.91E 00	9.96E 00	9.98E 00	9.98E 00
CR	5.38E 04	5.38E 04	5.38E 04	5.38E 04	5.38E 04	5.38E 04	5.38E 04
MN	5.98E 03	5.98E 03	5.97E 03	5.97E 03	5.97E 03	5.97E 03	5.99E 03
FE	2.12E 05	2.12E 05	2.12E 05	2.12E 05	2.12E 05	2.12E 05	2.12E 05
CO	2.99E 02	3.14E 02	3.10E 02	3.06E 02	3.04E 02	3.01E 02	3.00E 02
NI	2.39E 04	2.39E 04	2.39E 04	2.39E 04	2.39E 04	2.39E 04	2.39E 04
CU	0.0	2.82E-01	2.83E-01	2.84E-01	2.86E-01	3.07E-01	5.04E-01
TOTALS	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05	4.35E 05

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

NUCLIDE RADIOACTIVITY, CURIES / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30.0 D	90.0 D	150. D	1096. D	10958. D
C 14	0.0	5.84E-02	5.84E-02	5.84E-02	5.84E-02	5.83E-02	5.81E-02
P 32	0.0	2.31E 03	5.39E 02	2.94E 01	1.61E 00	0.0	0.0
P 33	0.0	1.35E 01	5.86E 00	1.11E 00	2.10E-01	8.49E-13	0.0
V 49	0.0	5.73E-02	5.38E-02	4.75E-02	4.18E-02	5.74E-03	5.79E-12
CR 51	0.0	6.17E 04	2.92E 04	6.55E 03	1.47E 03	8.37E-08	0.0
MN 54	0.0	1.50E 05	1.40E 05	1.22E 05	1.07E 05	1.22E 04	1.96E-06
FE 55	0.0	7.60E 04	7.43E 04	7.12E 04	6.81E 04	3.41E 04	2.55E 01
FE 59	0.0	1.03E 04	6.52E 03	2.59E 03	1.03E 03	4.82E-04	0.0
CO 58	0.0	4.10E 05	3.07E 05	1.71E 05	9.55E 04	1.00E 01	0.0
CO 60	0.0	1.35E 03	1.33E 03	1.31E 03	1.28E 03	9.08E 02	2.59E 01
NI 59	0.0	2.13E 00	2.13E 00	2.13E 00	2.13E 00	2.13E 00	2.13E 00
NI 63	0.0	6.78E 01	6.77E 01	6.76E 01	6.76E 01	6.63E 01	5.41E 01
SUBTOT	0.0	7.13E 05	5.59E 05	3.75E 05	2.74E 05	4.74E 04	1.08E 02
TOTALS	0.0	1.27E 06	5.59E 05	3.75E 05	2.74E 05	4.74E 04	1.08E 02

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

NUCLIDE THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30.0 D	90.0 D	150. D	1096. D	10958. D
P 32	0.0	9.51E 00	2.22E 00	1.21E-01	6.61E-03	0.0	0.0
CR 51	0.0	2.74E 02	1.30E 02	2.91E 01	6.51E 00	3.72E-10	0.0
MN 54	0.0	1.21E 03	1.13E 03	9.86E 02	8.60E 02	9.88E 01	1.58E-08
FE 55	0.0	9.91E 01	9.70E 01	9.28E 01	8.88E 01	4.45E 01	3.33E-02
FE 59	0.0	8.01E 01	5.05E 01	2.00E 01	7.95E 00	3.73E-06	0.0
CO 58	0.0	7.69E 03	5.74E 03	3.20E 03	1.79E 03	1.88E-01	0.0
CO 60	0.0	2.10E 01	2.08E 01	2.03E 01	1.99E 01	1.42E 01	4.03E-01
NI 63	0.0	1.08E-02	1.08E-02	1.08E-02	1.08E-02	1.06E-02	8.65E-03
SUBTOT	0.0	9.38E 03	7.17E 03	4.35E 03	2.77E 03	1.58E 02	4.45E-01
TOTALS	0.0	1.83E 04	7.17E 03	4.35E 03	2.77E 03	1.58E 02	4.45E-01

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

NUCLIDE GAMMA POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30.0 D	90.0 D	150. D	1096. D	10958. D
CR 51	0.0	1.18E 01	5.58E 00	1.25E 00	2.80E-01	1.60E-11	0.0
MN 54	0.0	7.39E 02	6.90E 02	6.02E 02	5.25E 02	6.02E 01	9.65E-09
FE 59	0.0	7.27E 01	4.58E 01	1.82E 01	7.21E 00	3.39E-06	0.0
CO 58	0.0	4.46E 03	3.33E 03	1.86E 03	1.04E 03	1.09E-01	0.0
CO 60	0.0	2.00E 01	1.98E 01	1.93E 01	1.89E 01	1.34E 01	3.83E-01
SUBTOT	0.0	5.30E 03	4.09E 03	2.50E 03	1.59E 03	7.38E 01	3.83E-01
TOTALS	0.0	1.08E 04	4.09E 03	2.50E 03	1.59E 03	7.38E 01	3.83E-01

CLADDING ACTIVATION IN AI LMFBR AT 80000 MWD/TONNE, 148.1 MW/TONNE

ELEMENT THERMAL POWER, WATTS / METRIC TON FUEL CHARGED TO REACTOR

	CHARGE	DISCHARGE	30. D	90. D	150. D	1096. D	10958. D
H	0.0	1.17E-09	1.16E-09	1.15E-09	1.14E-09	9.88E-10	2.16E-10
C	0.0	1.73E-05	1.73E-05	1.73E-05	1.73E-05	1.73E-05	1.72E-05
NA	0.0	4.22E 02	1.50E-12	0.0	0.0	0.0	0.0
P	0.0	9.54E 00	2.22E 00	1.22E-01	6.71E-03	3.83E-16	0.0
S	0.0	1.14E-07	9.03E-08	5.63E-08	3.51E-08	2.04E-11	0.0
CL	0.0	3.58E-14	8.24E-32	8.24E-32	8.24E-32	8.24E-32	8.24E-32
AR	0.0	9.53E-10	4.79E-22	1.46E-22	4.45E-23	0.0	0.0
K	0.0	3.53E-09	0.0	0.0	0.0	0.0	0.0
CA	0.0	5.27E-06	5.49E-08	1.12E-09	8.65E-10	1.63E-11	2.35E-26
SC	0.0	1.00E-02	1.78E-05	5.41E-09	3.22E-09	1.31E-12	0.0
V	0.0	3.91E 03	1.98E-04	1.74E-04	1.54E-04	2.11E-05	2.13E-14
CR	0.0	2.98E 02	1.30E 02	2.91E 01	6.51E 00	3.72E-10	0.0
MN	0.0	5.54E 03	1.13E 03	9.86E 02	8.60E 02	9.88E 01	1.58E-08
FE	0.0	1.79E 02	1.47E 02	1.13E 02	9.68E 01	4.45E 01	3.33E-02
CO	0.0	7.71E 03	5.76E 03	3.23E 03	1.81E 03	1.43E 01	4.03E-01
NI	0.0	1.14E 01	1.08E-02	1.08E-02	1.08E-02	1.06E-02	8.65E-03
CU	0.0	4.86E-02	1.51E-20	0.0	0.0	0.0	0.0
TOTALS	0.0	1.83E 04	7.17E 03	4.35E 03	2.77E 03	1.58E 02	4.45E-01

APPENDIX D: NUCLIDE DATA LIBRARY FOR AI REFERENCE OXIDE LMFBR

Symbols:

1. NUCL = NUCLIDE = 10000 * ATOMIC NO + 10 * MASS NO + ISOMERIC STATE (0 or 1)
2. DLAM = DECAY CONSTANT (1/SEC).
3. FB, FP, FA, FT = FRACTIONAL DECAY BY BETA, POSITRON (OR ELECTRON CAPTURE), ALPHA, INTERNAL TRANSITION. FB = 1 - FP - FA - FT
4. FB1, FP1, FNG1, FN2N1 = FRACTION OF BETA, POSITRON, N-GAMMA, N-2N TRANSITIONS TO EXCITED STATE OF PRODUCT NUCLIDE
5. SIGTH, SIGNG, SIGF, SIGNA, SIGNP, SIGN2N, SIGN3N = CROSS SECTIONS (BARNS) FOR ABSORPTION, N-GAMMA, FISSION, N-ALPHA, N-PROTON, N-2N, N-3N
6. Y25, Y28, Y49 = FISSION YIELD (PERCENT) FROM 233-U, 238-U, 239-Pu.
7. Q = HEAT PER DISINTEGRATION. FG = FRACTION OF HEAT IN GAMMAS OF ENERGY GREATER THAN 0.2 MEV.

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LIGHT ELEMENT LIBRARY FOR AI REFERENCE OXIDE LMFBR

LIGHT ELEMENTS, MATERIALS OF CONSTRUCTION, AND ACTIVATION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	FA	SIGNG	FNG1	SIGN2N	FN2N1	SIGNA	SIGNP	Q	FG	ABUNDANCE
H 1	0.0	0.0	0.0	0.0	0.0	0.0	2.00E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	99.985
H 2	0.0	0.0	0.0	0.0	0.0	0.0	1.80E-06	0.0	9.18E-04	0.0	0.0	0.0	0.0	0.0	0.015
H 3	1.79E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.006	0.0	0.0
HE 3	0.0	0.0	0.0	0.0	0.0	0.0	5.00E-06	0.0	0.0	0.0	0.0	3.98E 00	0.0	0.0	0.000
HE 4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	100.000
HE 6	8.56E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.580	0.0	0.0
LI 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.40E-04	0.0	6.22E-02	4.53E-04	0.0	0.0	7.420
LI 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.70E-05	0.0	5.53E-03	9.61E-06	0.0	0.0	92.580
LI 8	8.25E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	8.490	0.0	0.0
BE 8	6.93E 03	0.0	0.0	0.0	0.0	1.000	0.0	0.0	3.06E-04	0.0	5.00E-04	3.60E-08	0.095	0.0	0.0
BE 9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.80E-02	0.0	8.25E-03	6.00E-06	0.0	0.0	100.000
BE 10	8.79E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.60E-03	0.0	5.00E-04	3.60E-05	0.0	0.0	0.0
BE 11	5.10E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.25E-02	0.0	1.20E-03	5.00E-06	5.000	0.200	0.0
B 10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.04E 00	0.0	0.0	0.0	19.780
B 11	0.0	0.0	0.0	0.0	0.0	0.0	1.10E-02	0.0	7.70E-07	0.0	1.07E-05	1.60E-07	0.0	0.0	80.220
B 12	3.47E 01	0.0	0.0	0.0	0.0	0.0	3.30E-02	0.0	0.0	0.0	4.10E-05	4.80E-07	6.400	0.012	0.0
C 12	0.0	0.0	0.0	0.0	0.0	0.0	1.56E-03	0.0	0.0	0.0	4.95E-05	2.62E-07	0.0	0.0	98.890
C 13	0.0	0.0	0.0	0.0	0.0	0.0	4.13E-03	0.0	1.76E-03	0.0	2.86E-04	1.10E-08	0.0	0.0	1.110
C 14	3.83E-12	0.0	0.0	0.0	0.0	0.0	1.00E-03	0.0	0.0	0.0	5.30E-04	7.00E-06	0.050	0.0	0.0
N 13	1.16E-03	0.0	1.000	0.0	0.0	0.0	0.0	0.0	2.20E-06	0.0	6.00E-03	2.20E-06	1.490	0.680	0.0
N 14	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.10E-07	0.0	1.22E-02	0.0	0.0	0.0	99.630
N 15	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.20E-06	0.0	1.43E-06	2.20E-07	0.0	0.0	0.370
N 16	9.76E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.00E-06	0.0	9.75E-03	0.0	7.520	0.660	0.0
O 16	0.0	0.0	0.0	0.0	0.0	0.0	6.00E-04	0.0	1.10E-08	0.0	6.60E-04	1.87E-06	0.0	0.0	99.759
O 17	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.23E-03	0.0	0.0	2.20E-07	0.0	0.0	0.037
O 18	0.0	0.0	0.0	0.0	0.0	0.0	6.00E-04	0.0	7.26E-05	0.0	1.65E-04	0.0	0.0	0.0	0.204
O 19	2.39E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.10E-03	0.0	2.720	0.420	0.0
F 19	0.0	0.0	0.0	0.0	0.0	0.0	2.00E-03	0.0	2.57E-06	0.0	4.35E-03	3.61E-04	0.0	0.0	100.000
F 20	6.30E-02	0.0	0.0	0.0	0.0	0.0	6.00E-03	0.0	7.90E-05	0.0	7.20E-03	1.20E-04	4.130	0.390	0.0
NE 20	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.10E-08	0.0	5.50E-03	1.10E-05	0.0	0.0	90.920
NE 21	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.63E-04	0.0	4.40E-03	7.70E-05	0.0	0.0	0.257
NE 22	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.60E-06	0.0	8.80E-06	0.0	0.0	0.0	8.820
NE 23	1.84E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.76E-05	0.0	2.060	0.078	0.0
NA 23	0.0	0.0	0.0	0.0	0.0	0.0	1.70E-03	0.0	4.50E-07	0.0	8.26E-05	2.41E-04	0.0	0.0	100.000
NA 24	1.28E-05	0.0	0.0	0.0	0.0	0.0	5.10E-03	0.0	0.0	0.0	2.47E-04	1.00E-04	4.720	0.872	0.0
NA 25	1.16E-02	0.0	0.0	0.0	0.0	0.0	1.40E-03	0.0	0.0	0.0	6.60E-05	2.00E-05	1.910	0.199	0.0
MG 24	0.0	0.0	0.0	0.0	0.0	0.0	6.10E-03	0.0	0.0	0.0	1.90E-04	3.16E-04	0.0	0.0	78.700
MG 25	0.0	0.0	0.0	0.0	0.0	0.0	1.80E-02	0.0	2.20E-04	0.0	1.87E-03	2.04E-04	0.0	0.0	10.130
MG 26	0.0	0.0	0.0	0.0	0.0	0.0	8.20E-03	0.0	3.30E-06	0.0	3.79E-06	2.47E-05	0.0	0.0	11.170
MG 27	1.22E-03	0.0	0.0	0.0	0.0	0.0	2.40E-02	0.0	0.0	0.0	3.00E-06	2.00E-05	2.070	0.432	0.0
AL 27	0.0	0.0	0.0	0.0	0.0	0.0	2.00E-03	0.0	3.95E-07	0.0	1.60E-04	8.14E-04	0.0	0.0	100.000
AL 28	5.00E-03	0.0	0.0	0.0	0.0	0.0	6.00E-03	0.0	0.0	0.0	5.00E-04	3.00E-04	3.030	0.590	0.0
AL 29	1.75E-03	0.0	0.0	0.0	0.0	0.0	1.60E-03	0.0	0.0	0.0	1.00E-04	3.00E-04	2.340	0.577	0.0
SI 28	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.10E-08	0.0	9.60E-05	4.51E-04	0.0	0.0	99.280
SI 29	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	7.70E-05	0.0	4.17E-03	3.26E-04	0.0	0.0	4.700
SI 30	0.0	0.0	0.0	0.0	0.0	0.0	5.50E-03	0.0	7.70E-06	0.0	1.10E-05	2.81E-05	0.0	0.0	3.090
SI 31	7.35E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.594	0.001	0.0
P 31	0.0	0.0	0.0	0.0	0.0	0.0	1.70E-04	0.0	4.34E-07	0.0	1.46E-04	5.18E-03	0.0	0.0	100.000
P 32	5.61E-07	0.0	0.0	0.0	0.0	0.0	5.10E-04	0.0	0.0	0.0	0.0	0.0	0.695	0.0	0.0
P 33	3.21E-07	0.0	0.0	0.0	0.0	0.0	1.40E-04	0.0	0.0	0.0	0.0	0.0	0.076	0.0	0.0

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NUCL	DLAM	FB1	FP	FP1	FT	FA	SIGNG	FNG1	SIGN2N	FN2N1	SIGNA	SIGNP	Q	FG	ABUNDANCE
P 34	5.59E-02	0.0	0.0	0.0	0.0	0.0	4.00E-04	0.0	0.0	0.0	0.0	0.0	4.160	0.128	0.0
S 32	0.0	0.0	0.0	0.0	0.0	0.0	1.90E-04	0.0	5.50E-08	0.0	6.14E-03	9.43E-03	0.0	0.0	95.000
S 33	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.50E-05	0.0	7.26E-03	7.15E-03	0.0	0.0	0.760
S 34	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.30E-06	0.0	2.75E-04	6.93E-05	0.0	0.0	4.220
S 35	9.12E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.048	0.0	0.0
S 36	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.87E-05	0.0	3.85E-06	2.20E-08	0.0	0.0	0.014
S 37	2.28E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.560	0.780	0.0
CL 35	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.20E-07	0.0	1.54E-03	2.15E-03	0.0	0.0	75.730
CL 36	7.09E-14	0.0	0.0	0.019	0.0	0.0	0.0	0.0	8.25E-05	0.0	2.75E-03	0.0	0.314	0.0	0.0
CL 37	0.0	0.0	0.0	0.0	0.0	0.0	4.30E-03	0.0	1.21E-05	0.0	1.07E-04	4.00E-05	0.0	0.0	24.470
CL 38	3.10E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.100	0.525	0.0
AR 36	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.90E-08	0.0	8.42E-03	2.00E-02	0.0	0.0	0.337
AR 37	2.29E-07	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.814	0.0	0.0
AR 38	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.20E-06	0.0	2.42E-04	7.70E-05	0.0	0.0	0.063
AR 39	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.234	0.0	0.0
AR 40	0.0	0.0	0.0	0.0	0.0	0.0	1.90E-03	0.0	2.42E-05	0.0	6.25E-06	9.90E-07	0.0	0.0	99.600
AR 41	1.05E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.590	0.200	0.0
K 39	0.0	0.0	0.0	0.0	0.0	0.0	2.00E-03	0.0	1.90E-07	0.0	1.40E-03	2.01E-02	0.0	0.0	93.100
K 40	1.69E-17	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.09E-04	0.0	5.17E-03	0.0	0.0	0.0	0.012
K 41	0.0	0.0	0.0	0.0	0.0	0.0	1.00E-03	0.0	1.65E-05	0.0	3.92E-04	3.32E-04	0.0	0.0	6.880
K 42	1.55E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.760	0.204	0.0
K 43	8.60E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.290	0.763	0.0
K 44	5.25E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.300	0.534	0.0
CA 40	0.0	0.0	0.0	0.0	0.0	0.0	1.90E-03	0.0	3.30E-08	0.0	7.69E-04	5.53E-03	0.0	0.0	96.970
CA 41	2.75E-13	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.427	0.0	0.0
CA 42	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.40E-06	0.0	2.64E-04	3.33E-04	0.0	0.0	0.640
CA 43	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.98E-04	0.0	9.24E-04	3.74E-04	0.0	0.0	0.145
CA 44	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.50E-06	0.0	8.46E-06	2.46E-05	0.0	0.0	2.060
CA 45	4.86E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.103	0.0	0.0
CA 46	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.08E-05	0.0	2.20E-08	9.90E-08	0.0	0.0	0.003
CA 47	1.77E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.460	0.744	0.0
CA 48	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.36E-05	0.0	0.0	0.0	0.0	0.0	0.180
CA 49	1.31E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.070	0.783	0.0
SC 45	0.0	0.0	0.0	0.0	0.0	0.0	2.78E-02	0.0	9.97E-06	0.0	5.98E-05	2.43E-03	0.0	0.0	100.000
SC 46	9.56E-08	0.0	0.0	0.0	0.0	0.0	1.67E-01	0.0	0.0	0.0	0.0	0.0	2.370	0.849	0.0
SC 47	2.34E-06	0.0	0.0	0.0	0.0	0.0	7.00E-02	0.0	0.0	0.0	0.0	0.0	1.460	0.745	0.0
SC 48	4.38E-06	0.0	0.0	0.0	0.0	0.0	2.60E-01	0.0	0.0	0.0	0.0	0.0	3.570	0.939	0.0
SC 49	2.01E-04	0.0	0.0	0.0	0.0	0.0	1.10E-01	0.0	0.0	0.0	0.0	0.0	0.825	0.0	0.0
SC 50	6.72E-03	0.0	0.0	0.0	0.0	0.0	4.20E-01	0.0	0.0	0.0	0.0	0.0	4.640	0.689	0.0
TI 46	0.0	0.0	0.0	0.0	0.0	0.0	1.54E-02	0.0	9.56E-07	0.0	7.26E-05	1.30E-03	0.0	0.0	7.930
TI 47	0.0	0.0	0.0	0.0	0.0	0.0	2.80E-02	0.0	8.00E-05	0.0	6.05E-04	2.39E-03	0.0	0.0	7.280
TI 48	0.0	0.0	0.0	0.0	0.0	0.0	5.36E-03	0.0	4.40E-06	0.0	1.68E-05	1.35E-04	0.0	0.0	73.940
TI 49	0.0	0.0	0.0	0.0	0.0	0.0	9.74E-03	0.0	1.76E-04	0.0	8.72E-05	3.49E-04	0.0	0.0	5.510
TI 50	0.0	0.0	0.0	0.0	0.0	0.0	2.03E-03	0.0	7.70E-06	0.0	8.81E-07	1.45E-05	0.0	0.0	5.340
TI 51	1.99E-03	0.0	0.0	0.0	0.0	0.0	3.65E-03	0.0	0.0	0.0	0.0	0.0	1.180	0.260	0.0
V 49	2.43E-08	0.0	1.000	0.0	0.0	0.0	3.44E-03	0.0	0.0	0.0	0.0	0.0	0.620	0.0	0.0
V 50	0.0	0.0	0.0	0.0	0.0	0.0	1.30E-02	0.0	5.50E-05	0.0	2.31E-04	0.0	0.0	0.0	0.240
V 51	0.0	0.0	0.0	0.0	0.0	0.0	5.39E-03	0.0	2.33E-05	0.0	6.60E-06	1.26E-04	0.0	0.0	99.760
V 52	3.43E-03	0.0	0.0	0.0	0.0	0.0	2.06E-02	0.0	0.0	0.0	0.0	0.0	2.610	0.550	0.0
V 53	5.78E-03	0.0	0.0	0.0	0.0	0.0	8.50E-03	0.0	0.0	0.0	0.0	0.0	1.890	0.528	0.0

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NUCL	DLAM	FB1	FP	FP1	FT	FA	SIGNG	FNG1	SIGN2N	FN2N1	SIGNA	SIGNP	Q	FG	ABUNDANCE
V 54	1.26E-02	0.0	0.0	0.0	0.0	0.0	3.16E-02	0.0	0.0	0.0	0.0	0.0	5.460	0.740	0.0
CR 50	0.0	0.0	0.0	0.0	0.0	0.0	1.25E-02	0.0	4.69E-07	0.0	1.05E-04	2.85E-03	0.0	0.0	4.310
CR 51	2.89E-07	0.0	1.000	0.0	0.0	0.0	4.04E-01	0.0	0.0	0.0	0.0	0.0	0.749	0.043	0.0
CR 52	0.0	0.0	0.0	0.0	0.0	0.0	1.53E-03	0.0	9.05E-06	0.0	8.03E-06	3.19E-03	0.0	0.0	83.760
CR 53	0.0	0.0	0.0	0.0	0.0	0.0	1.62E-01	0.0	2.97E-04	0.0	3.30E-04	1.62E-04	0.0	0.0	9.550
CR 54	0.0	0.0	0.0	0.0	0.0	0.0	6.50E-04	0.0	1.21E-04	0.0	5.06E-06	4.40E-07	0.0	0.0	2.380
CR 55	3.30E-03	0.0	0.0	0.0	0.0	0.0	1.78E-02	0.0	0.0	0.0	0.0	0.0	1.230	0.0	0.0
MN 54	2.65E-08	0.0	1.000	0.0	0.0	0.0	4.63E-02	0.0	0.0	0.0	0.0	0.0	1.360	0.610	0.0
MN 55	0.0	0.0	0.0	0.0	0.0	0.0	1.93E-02	0.0	6.56E-05	0.0	3.42E-04	1.69E-04	0.0	0.0	100.000
MN 56	7.46E-05	0.0	0.0	0.0	0.0	0.0	7.17E-02	0.0	0.0	0.0	0.0	0.0	2.700	0.670	0.0
MN 57	6.80E-03	0.0	0.0	0.0	0.0	0.0	3.06E-02	0.0	0.0	0.0	0.0	0.0	1.530	0.460	0.0
MN 58	1.05E-02	0.0	0.0	0.0	0.0	0.0	1.17E-01	0.0	0.0	0.0	0.0	0.0	3.820	0.438	0.0
FE 54	0.0	0.0	0.0	0.0	0.0	0.0	1.21E-02	0.0	6.81E-07	0.0	5.92E-04	1.13E-02	0.0	0.0	5.820
FE 55	8.45E-09	0.0	1.000	0.0	0.0	0.0	3.86E-02	0.0	0.0	0.0	0.0	0.0	0.220	0.0	0.0
FE 56	0.0	0.0	0.0	0.0	0.0	0.0	1.01E-02	0.0	7.70E-06	0.0	3.85E-05	2.63E-04	0.0	0.0	91.660
FE 57	0.0	0.0	0.0	0.0	0.0	0.0	3.22E-02	0.0	3.30E-04	0.0	3.85E-04	9.77E-05	0.0	0.0	2.190
FE 58	0.0	0.0	0.0	0.0	0.0	0.0	8.32E-03	0.0	2.20E-05	0.0	5.17E-06	6.97E-05	0.0	0.0	0.330
FE 59	1.78E-07	0.0	0.0	0.0	0.0	0.0	2.69E-02	0.0	0.0	0.0	0.0	0.0	1.306	0.908	0.0
CO 58M	2.14E-05	0.0	0.0	0.0	1.000	0.0	6.02E-02	0.0	0.0	0.0	0.0	0.0	0.025	0.0	0.0
CO 58	1.13E-07	0.0	1.000	0.0	0.0	0.0	3.01E-02	0.0	0.0	0.0	0.0	0.0	3.160	0.580	0.0
CO 59	0.0	0.0	0.0	0.0	0.0	0.0	1.20E-02	0.0	3.61E-05	0.0	4.00E-05	2.84E-04	0.0	0.0	100.000
CO 60M	1.10E-03	0.0	0.0	0.0	0.997	0.0	8.95E-02	0.0	2.60E-04	0.0	1.44E-04	3.16E-04	0.063	0.063	0.0
CO 60	4.18E-09	0.0	0.0	0.0	0.0	0.0	4.45E-02	0.0	1.30E-04	0.0	7.20E-05	1.58E-04	2.630	0.950	0.0
CO 61	1.17E-04	0.0	0.0	0.0	0.0	0.0	1.89E-02	0.0	0.0	0.0	0.0	0.0	0.556	0.0	0.0
CO 62	8.31E-04	0.0	0.0	0.0	0.0	0.0	7.34E-02	0.0	0.0	0.0	0.0	0.0	3.850	0.740	0.0
NI 58	0.0	0.0	0.0	0.0	0.0	0.0	7.21E-03	0.0	6.85E-07	0.0	3.10E-04	1.70E-02	0.0	0.0	67.880
NI 59	2.75E-13	0.0	1.000	0.0	0.0	0.0	2.04E-02	0.0	0.0	0.0	2.20E-03	1.32E-03	0.0	0.0	0.0
NI 60	0.0	0.0	0.0	0.0	0.0	0.0	6.01E-03	0.0	7.70E-06	0.0	1.41E-04	2.83E-04	0.0	0.0	26.230
NI 61	0.0	0.0	0.0	0.0	0.0	0.0	1.72E-02	0.0	1.43E-04	0.0	5.02E-04	2.86E-04	0.0	0.0	1.190
NI 62	0.0	0.0	0.0	0.0	0.0	0.0	4.89E-03	0.0	1.98E-05	0.0	7.63E-06	2.40E-05	0.0	0.0	3.660
NI 63	2.39E-10	0.0	0.0	0.0	0.0	0.0	1.44E-02	0.0	0.0	0.0	0.0	0.0	0.027	0.0	0.0
NI 64	0.0	0.0	0.0	0.0	0.0	0.0	4.00E-03	0.0	4.95E-05	0.0	4.34E-07	1.42E-06	0.0	0.0	1.080
NI 65	7.52E-05	0.0	0.0	0.0	0.0	0.0	1.20E-02	0.0	0.0	0.0	0.0	0.0	1.210	0.490	0.0
CU 62	1.18E-03	0.0	1.000	0.0	0.0	0.0	7.31E-02	0.0	0.0	0.0	0.0	0.0	2.330	0.440	0.0
CU 63	0.0	0.0	0.0	0.0	0.0	0.0	2.20E-02	0.0	2.12E-05	0.0	8.71E-05	1.36E-03	0.0	0.0	69.090
CU 64	1.50E-05	0.0	0.620	0.0	0.0	0.0	6.00E-02	0.0	0.0	0.0	0.0	0.0	1.030	0.190	0.0
CU 65	0.0	0.0	0.0	0.0	0.0	0.0	1.80E-02	0.0	7.68E-05	0.0	1.58E-05	6.20E-05	0.0	0.0	30.910
CU 66	2.27E-03	0.0	0.0	0.0	0.0	0.0	4.84E-02	0.0	0.0	0.0	0.0	0.0	1.160	0.080	0.0
NB 92	7.90E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.010	0.465	0.0
NB 93M	1.62E-09	0.0	0.0	0.0	1.000	0.0	1.08E-01	0.0	7.09E-05	0.0	6.58E-05	1.54E-04	0.030	0.0	0.0
NB 93	0.0	0.0	0.0	0.0	0.0	0.0	1.08E-01	0.0	7.09E-05	0.0	6.58E-05	1.54E-04	0.0	0.0	100.000
NB 94	1.10E-15	0.0	0.0	0.0	0.0	0.0	8.46E-01	0.0	8.25E-04	0.0	2.31E-04	5.50E-03	1.970	0.799	0.0
NB 95	2.29E-07	0.0	0.0	0.0	0.0	0.0	3.93E-01	0.0	0.0	0.0	0.0	0.0	0.809	0.945	0.0
NB 96	8.37E-06	0.0	0.0	0.0	0.0	0.0	1.79E 00	0.0	0.0	0.0	0.0	0.0	2.610	0.900	0.0
NB 97	1.60E-04	0.0	0.0	0.0	0.0	0.0	5.11E-01	0.0	0.0	0.0	0.0	0.0	1.160	0.580	0.0
MO 92	0.0	0.0	0.0	0.0	0.0	0.0	2.83E-02	0.0	4.02E-06	0.0	5.93E-06	3.44E-04	0.0	0.0	15.840
MO 93M	2.79E-05	0.0	0.0	0.0	0.0	0.0	2.44E-02	0.0	0.0	0.0	0.0	0.0	2.470	1.000	0.0
MO 93	2.44E-12	0.0	1.000	0.850	0.0	0.0	1.22E-02	0.0	0.0	0.0	0.0	0.0	0.420	0.0	0.0
MO 94	0.0	0.0	0.0	0.0	0.0	0.0	3.28E-02	0.0	5.50E-05	0.0	7.26E-05	9.13E-05	0.0	0.0	9.040
MO 95	0.0	0.0	0.0	0.0	0.0	0.0	1.41E-01	0.0	9.13E-04	0.0	4.07E-04	5.17E-05	0.0	0.0	15.720

LIGHT ELEMENT LIBRARY FOR AI REFERENCE OXIDE LMFBR

LIGHT ELEMENTS, MATERIALS OF CONSTRUCTION, AND ACTIVATION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	FA	SIGNG	FNG1	SIGN2N	FN2N1	SIGNA	SIGNP	Q	FG	ABUNDANCE
MO 96	0.0	0.0	0.0	0.0	0.0	0.0	3.73E-02	0.0	1.01E-04	0.0	3.22E-05	2.62E-05	0.0	0.0	16.530
MO 97	0.0	0.0	0.0	0.0	0.0	0.0	1.61E-01	0.0	1.10E-03	0.0	1.46E-04	8.39E-05	0.0	0.0	9.460
MO 98	0.0	0.0	0.0	0.0	0.0	0.0	4.32E-02	0.0	2.53E-04	0.0	7.15E-06	1.02E-06	0.0	0.0	23.780
MO 99	2.87E-06	0.0	0.0	0.0	0.0	0.0	1.84E-01	0.0	0.0	0.0	0.0	0.0	0.758	0.290	0.0
MO100	0.0	0.0	0.0	0.0	0.0	0.0	4.91E-02	0.0	5.31E-04	0.0	2.66E-06	2.39E-07	0.0	0.0	9.630
MO101	7.91E-04	0.0	0.0	0.0	0.0	0.0	2.08E-01	0.0	0.0	0.0	0.0	0.0	1.950	0.680	0.0
TC 99M	3.21E-05	0.0	0.0	0.0	0.0	0.0	2.04E-01	0.0	1.10E-04	0.0	3.43E-06	1.04E-05	0.143	0.0	0.0
TC 99	1.05E-13	0.0	0.0	0.0	0.0	0.0	2.04E-01	0.0	1.10E-04	0.0	3.43E-06	1.04E-05	0.290	0.0	0.0
TC101	8.25E-04	0.0	0.0	0.0	0.0	0.0	1.98E-01	0.0	0.0	0.0	0.0	0.0	0.803	0.410	0.0
RU101	0.0	0.0	0.0	0.0	0.0	0.0	5.11E-01	0.0	1.16E-03	0.0	1.21E-04	5.18E-05	0.0	0.0	17.070
W180	0.0	0.0	0.0	0.0	0.0	0.0	1.79E-01	0.0	2.97E-04	0.0	6.60E-08	9.90E-06	0.0	0.0	0.140
W181	5.73E-08	0.0	1.000	0.0	0.0	0.0	3.54E-01	0.0	0.0	0.0	0.0	0.0	0.170	0.0	0.0
W182	0.0	0.0	0.0	0.0	0.0	0.0	9.86E-02	0.0	4.51E-04	0.0	1.10E-08	2.20E-06	0.0	0.0	26.410
W183M	1.31E-01	0.0	0.0	0.0	1.000	0.0	2.18E-01	0.0	2.09E-03	0.0	1.10E-08	8.80E-07	0.260	0.0	0.0
W183	0.0	0.0	0.0	0.0	0.0	0.0	2.18E-01	0.0	2.09E-03	0.0	1.10E-08	8.80E-07	0.0	0.0	14.400
W184	0.0	0.0	0.0	0.0	0.0	0.0	6.84E-02	0.0	7.41E-04	0.0	7.38E-08	5.77E-07	0.0	0.0	30.640
W185M	7.22E-03	0.0	0.0	0.0	1.000	0.0	2.13E-01	0.0	0.0	0.0	0.0	0.0	0.290	0.0	0.0
W185	1.07E-07	0.0	0.0	0.0	0.0	0.0	2.13E-01	0.0	0.0	0.0	0.0	0.0	0.160	0.0	0.0
W186	0.0	0.0	0.0	0.0	0.0	0.0	9.26E-02	0.0	8.43E-04	0.0	1.88E-08	7.01E-07	0.0	0.0	28.410
W187	9.06E-06	0.0	0.0	0.0	0.0	0.0	3.16E-01	0.0	0.0	0.0	0.0	0.0	0.865	0.670	0.0
PB206	0.0	0.0	0.0	0.0	0.0	0.0	3.40E-03	0.0	3.96E-04	0.0	0.0	1.10E-06	0.0	0.0	23.600
PB209	5.83E-05	0.0	0.0	0.0	0.0	0.0	1.99E-03	0.0	0.0	0.0	0.0	0.0	0.195	0.0	0.0
BI208	0.0	0.0	0.0	0.0	0.0	0.0	1.85E-02	0.0	1.87E-03	0.0	1.10E-08	3.85E-04	0.0	0.0	0.0
BI209	0.0	0.0	0.0	0.0	0.0	0.0	4.93E-03	0.0	4.65E-04	0.0	7.15E-08	5.68E-07	0.0	0.0	100.000
BI210M	0.0	0.0	0.0	0.0	0.0	0.0	2.67E-02	0.0	2.04E-02	0.0	1.10E-06	8.80E-06	0.0	0.0	0.0
BI210	1.60E-06	0.0	0.0	0.0	0.0	0.0	2.67E-02	0.0	2.04E-02	0.0	1.10E-06	8.80E-06	0.445	0.0	0.0
PO210	5.80E-06	0.0	0.0	0.0	0.0	1.000	2.30E-03	0.0	5.72E-04	0.0	2.00E-04	0.0	5.408	0.0	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
10030	1.79E-09	0.0	0.0	0.0	0.0	0.0	0.0	1.20E-02	1.40E-02	2.50E-02	0.006	0.0
300720	4.14E-06	0.0	0.0	0.0	0.0	3.15E-03	0.0	1.52E-03	1.00E-04	1.20E-03	0.246	0.084
310720	1.37E-05	0.0	0.0	0.0	0.0	1.45E-01	0.0	0.0	0.0	0.0	3.460	0.860
320720	0.0	0.0	0.0	0.0	0.0	2.81E-02	0.0	0.0	0.0	0.0	0.0	0.0
310730	3.93E-05	0.0	0.0	0.0	0.0	3.70E-02	0.0	1.90E-04	2.00E-04	0.0	0.673	0.0
320730	0.0	0.0	0.0	0.0	0.0	8.80E-02	0.0	0.0	0.0	0.0	0.0	0.0
310740	1.46E-03	0.0	0.0	0.0	0.0	1.12E-01	0.0	3.32E-02	7.00E-04	0.0	4.030	0.720
320740	0.0	0.0	0.0	0.0	0.0	1.78E-02	0.400	0.0	0.0	0.0	0.0	0.0
310750	5.73E-03	0.960	0.0	0.0	0.0	2.83E-02	0.0	3.79E-02	1.00E-03	0.0	2.030	0.285
320751	1.44E-02	0.0	0.0	0.0	1.000	5.39E-02	0.0	3.79E-02	0.0	0.0	0.140	0.0
320750	1.41E-04	0.0	0.0	0.0	0.0	5.39E-02	0.0	0.0	0.0	0.0	0.464	0.071
330750	0.0	0.0	0.0	0.0	0.0	1.98E-01	0.0	0.0	0.0	0.0	0.0	0.0
310760	2.17E-02	0.0	0.0	0.0	0.0	8.60E-02	0.0	9.49E-03	2.00E-03	0.0	3.280	0.170
320760	0.0	0.0	0.0	0.0	0.0	1.03E-02	0.500	9.49E-03	0.0	0.0	0.0	0.0
330760	7.27E-06	0.0	0.0	0.0	0.0	6.42E-01	0.0	0.0	0.0	0.0	2.370	0.180
340760	0.0	0.0	0.0	0.0	0.0	1.62E-01	0.259	0.0	0.0	0.0	0.0	0.0
320771	1.28E-02	0.0	0.0	0.0	0.240	3.25E-02	0.0	5.12E-02	2.60E-03	0.0	1.100	0.017
320770	1.70E-05	0.0	0.0	0.0	0.0	3.25E-02	0.0	3.51E-02	1.20E-03	0.0	1.770	0.660
330770	4.98E-06	0.003	0.0	0.0	0.0	1.62E-01	0.0	0.0	0.0	0.0	0.235	0.021
340771	3.96E-02	0.0	0.0	0.0	1.000	4.26E-01	0.0	0.0	0.0	0.0	0.161	0.0
340770	0.0	0.0	0.0	0.0	0.0	4.26E-01	0.0	0.0	0.0	0.0	0.0	0.0
320780	1.31E-04	0.0	0.0	0.0	0.0	6.34E-03	0.0	1.71E-01	1.60E-02	0.0	0.521	0.540
330781	1.93E-03	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.500	1.000
330780	1.27E-04	0.0	0.0	0.0	0.0	5.07E-01	0.0	1.90E-02	0.0	0.0	2.250	0.360
340780	0.0	0.0	0.0	0.0	0.0	7.13E-02	0.875	0.0	0.0	0.0	0.0	0.0
330790	1.28E-03	1.000	0.0	0.0	0.0	1.26E-01	0.0	3.79E-01	3.00E-02	0.0	0.0	0.0
340791	2.96E-03	0.0	0.0	0.0	1.000	1.82E-01	0.0	0.0	0.0	0.0	0.096	0.0
340790	3.38E-13	0.0	0.0	0.0	0.0	1.82E-01	0.0	0.0	0.0	0.0	0.064	0.0
350790	0.0	0.0	0.0	0.0	0.0	5.54E-01	0.254	0.0	3.00E-06	0.0	0.0	0.0
330800	4.53E-02	0.0	0.0	0.0	0.0	3.76E-01	0.0	1.52E-01	7.00E-02	0.0	3.090	0.210
340800	0.0	0.0	0.0	0.0	0.0	2.97E-02	0.200	0.0	0.0	0.0	0.0	0.0
350801	4.40E-05	0.0	0.0	0.0	1.000	1.17E 00	0.0	0.0	7.00E-06	0.0	0.086	0.0
350800	6.56E-04	0.0	0.080	0.0	0.0	1.17E 00	0.0	0.0	0.0	0.0	0.875	0.130
360800	0.0	0.0	0.0	0.0	0.0	3.04E-02	0.0	0.0	0.0	0.0	0.0	0.0
330810	2.10E-02	1.000	0.0	0.0	0.0	9.11E-02	0.0	2.37E-01	7.00E-03	0.0	1.670	0.0
340811	2.03E-04	0.0	0.0	0.0	1.000	7.52E-02	0.0	1.59E-02	0.0	0.0	0.103	0.0
340810	6.21E-04	0.0	0.0	0.0	0.0	7.52E-02	0.0	0.0	1.10E-01	0.0	0.828	0.340
350810	0.0	0.0	0.0	0.0	0.0	1.66E-01	0.0	0.0	0.0	0.0	0.0	0.0
360811	5.33E-02	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.190	0.0
360810	0.0	0.0	0.0	0.0	0.0	1.86E-01	0.0	0.0	0.0	0.0	0.0	0.0
340820	0.0	0.0	0.0	0.0	0.0	1.19E-02	0.920	7.21E-05	2.20E-01	0.0	0.0	0.0
350821	1.89E-03	0.0	0.0	0.0	1.000	0.0	0.0	0.0	3.00E-05	0.0	0.046	0.0
350820	5.45E-06	0.0	0.0	0.0	0.0	3.52E-01	0.0	0.0	0.0	0.0	2.787	0.950
360820	0.0	0.0	0.0	0.0	0.0	6.32E-02	1.000	0.0	0.0	0.0	0.0	0.0
340831	9.90E-03	0.0	0.0	0.0	0.0	2.93E-02	0.0	5.69E-01	3.50E-02	2.40E-01	1.680	0.0
340830	4.62E-04	0.0	0.0	0.0	0.0	2.93E-02	0.0	3.41E-01	2.60E-01	1.72E-01	2.670	0.750
350830	7.99E-05	1.000	0.0	0.0	0.0	6.92E-02	0.0	0.0	0.0	1.68E-01	0.387	0.0
360831	1.04E-04	0.0	0.0	0.0	1.000	2.17E-01	0.0	0.0	0.0	0.0	0.042	0.0
360830	0.0	0.0	0.0	0.0	0.0	2.17E-01	0.0	0.0	1.50E-01	0.0	0.0	0.0
340840	3.50E-03	0.0	0.0	0.0	0.0	4.55E-03	0.0	1.16E-01	7.80E-01	6.00E-02	1.280	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
350841	1.93E-03	0.0	0.0	0.0	0.0	1.94E-01	0.0	3.60E-02	1.60E-02	2.00E-02	3.660	0.790
350840	3.63E-04	0.0	0.0	0.0	0.0	1.94E-01	0.0	1.75E 00	0.0	8.60E-01	2.870	0.500
360840	0.0	0.0	0.0	0.0	0.0	3.36E-02	0.714	0.0	5.20E-02	0.0	0.0	0.0
340850	1.78E-02	0.0	0.0	0.0	0.0	1.12E-02	0.0	1.04E 00	4.27E-01	2.52E-01	3.200	0.0
350850	3.85E-03	1.000	0.0	0.0	0.0	4.67E-02	0.0	3.79E-01	0.0	1.60E-01	1.050	0.0
360851	4.38E-05	0.0	0.0	0.0	0.230	8.69E-02	0.0	0.0	2.20E-01	0.0	0.407	0.172
360850	2.04E-09	0.0	0.0	0.0	0.0	8.69E-02	0.0	0.0	8.90E-02	1.27E-01	0.271	0.008
370850	0.0	0.0	0.0	0.0	0.0	7.89E-02	0.0	0.0	0.0	0.0	0.0	0.0
350860	1.28E-02	0.0	0.0	0.0	0.0	1.50E-01	0.0	1.92E 00	1.38E 00	7.60E-01	3.230	0.0
360860	0.0	0.0	0.0	0.0	0.0	1.03E-02	0.0	0.0	0.0	0.0	0.0	0.0
370861	1.11E-02	0.0	0.0	0.0	1.000	1.80E-01	0.0	0.0	2.00E-06	0.0	0.560	1.000
370860	4.30E-07	0.0	0.0	0.0	0.0	1.80E-01	0.0	2.66E-05	0.0	2.30E-05	0.794	0.116
380860	0.0	0.0	0.0	0.0	0.0	3.00E-02	1.000	0.0	0.0	0.0	0.0	0.0
350870	1.26E-02	0.0	0.0	0.0	0.0	3.95E-02	0.0	2.56E 00	1.80E 00	9.20E-01	5.650	0.670
360870	1.52E-04	0.0	0.0	0.0	0.0	2.25E-02	0.0	0.0	0.0	0.0	2.430	0.440
370870	4.39E-19	0.0	0.0	0.0	0.0	2.76E-02	0.0	0.0	0.0	0.0	0.110	0.0
380871	6.80E-05	0.0	0.0	0.0	0.993	4.26E-02	0.0	0.0	0.0	0.0	0.388	1.000
380870	0.0	0.0	0.0	0.0	0.0	4.26E-02	0.0	0.0	0.0	0.0	0.0	0.0
350880	4.33E-02	0.0	0.0	0.0	0.0	1.27E-01	0.0	2.75E 00	2.50E 00	1.10E 00	4.050	0.0
360880	6.88E-05	0.0	0.0	0.0	0.0	5.14E-03	0.0	7.59E-01	0.0	3.20E-01	2.030	0.830
370880	6.49E-04	0.0	0.0	0.0	0.0	5.41E-02	0.0	0.0	0.0	0.0	2.690	0.210
380880	0.0	0.0	0.0	0.0	0.0	2.76E-03	0.0	0.0	0.0	0.0	0.0	0.0
350890	1.54E-01	0.0	0.0	0.0	0.0	3.40E-02	0.0	1.90E-01	2.78E 00	7.00E-02	3.270	0.0
360890	3.61E-03	0.0	0.0	0.0	0.0	1.42E-02	0.0	4.17E 00	0.0	1.64E 00	2.740	0.475
370890	7.50E-04	0.0	0.0	0.0	0.0	8.88E-03	0.0	1.90E-01	0.0	0.0	2.990	0.800
380890	1.54E-07	0.0	0.0	0.0	0.0	3.08E-03	0.0	0.0	1.20E-01	0.0	0.607	0.0
390890	0.0	0.0	0.0	0.0	0.0	6.89E-03	0.0	0.0	0.0	0.0	0.0	0.0
360900	2.10E-02	0.0	0.0	0.0	0.0	3.95E-03	0.0	4.93E 00	2.77E 00	1.96E 00	3.110	0.670
370900	3.98E-03	0.0	0.0	0.0	0.0	2.25E-02	0.0	6.64E-01	0.0	2.90E-01	4.700	0.690
380900	7.82E-10	0.0	0.0	0.0	0.0	1.58E-04	0.0	0.0	4.30E-01	0.0	0.220	0.0
390901	6.21E-05	0.0	0.0	0.0	1.000	3.94E-02	0.0	0.0	0.0	0.0	0.685	1.000
390900	3.01E-06	0.0	0.0	0.0	0.0	3.94E-02	0.0	0.0	0.0	0.0	0.965	0.0
400900	0.0	0.0	0.0	0.0	0.0	5.31E-03	0.0	0.0	0.0	0.0	0.0	0.0
360910	6.93E-02	0.0	0.0	0.0	0.0	1.11E-02	0.0	3.51E 00	2.38E 00	1.26E 00	1.570	0.0
370910	9.63E-03	0.0	0.0	0.0	0.0	4.93E-03	0.0	1.90E 00	1.36E 00	1.10E 00	2.950	0.305
380910	1.99E-05	0.590	0.0	0.0	0.0	1.38E-04	0.0	0.0	2.60E-01	0.0	1.740	0.620
390911	2.31E-04	0.0	0.0	0.0	1.000	1.65E-02	0.0	0.0	0.0	0.0	0.551	1.000
390910	1.36E-07	0.0	0.0	0.0	0.0	1.65E-02	0.0	0.0	0.0	0.0	0.642	0.006
400910	0.0	0.0	0.0	0.0	0.0	2.64E-02	0.0	0.0	4.00E-02	0.0	0.0	0.0
360920	2.31E-01	0.0	0.0	0.0	0.0	3.04E-03	0.0	2.56E 00	1.40E 00	1.53E 00	2.250	0.0
370920	1.31E-01	0.0	0.0	0.0	0.0	1.54E-02	0.0	2.66E 00	2.56E 00	1.54E 00	3.490	0.0
380920	7.10E-05	0.0	0.0	0.0	0.0	5.00E-06	0.0	5.69E-01	0.0	7.00E-02	1.460	0.840
390920	5.45E-05	0.0	0.0	0.0	0.0	8.08E-02	0.0	0.0	5.40E-01	0.0	1.780	0.210
400920	0.0	0.0	0.0	0.0	0.0	7.87E-03	0.0	0.0	0.0	0.0	0.0	0.0
360930	3.47E-01	0.0	0.0	0.0	0.0	8.30E-03	0.0	1.23E 00	3.70E-01	7.70E-01	3.680	0.0
370930	1.24E-01	0.0	0.0	0.0	0.0	3.95E-03	0.0	2.94E 00	4.35E 00	1.82E 00	2.750	0.0
380930	1.44E-03	0.0	0.0	0.0	0.0	4.00E-06	0.0	1.90E 00	0.0	1.32E 00	3.220	0.620
390930	1.89E-05	0.0	0.0	0.0	0.0	2.80E-02	0.0	9.49E-02	0.0	6.00E-02	1.380	0.200
400930	1.46E-14	1.000	0.0	0.0	0.0	3.94E-02	0.0	0.0	2.70E-01	0.0	0.020	0.0
410931	1.62E-09	0.0	0.0	0.0	1.000	1.08E-01	0.0	0.0	0.0	0.0	0.030	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
410930	0.0	0.0	0.0	0.0	0.0	1.08E-01	0.909	0.0	0.0	0.0	0.0	0.0
360940	4.95E-01	0.0	0.0	0.0	0.0	2.37E-03	0.0	5.69E-01	8.80E-02	6.00E-01	2.910	0.0
370940	2.39E-01	0.0	0.0	0.0	0.0	1.34E-02	0.0	2.18E 00	2.10E 00	1.50E 00	4.170	0.0
380940	8.89E-03	0.0	0.0	0.0	0.0	0.0	0.0	2.75E 00	2.54E 00	1.64E 00	2.250	0.630
390940	5.69E-04	0.0	0.0	0.0	0.0	1.24E-01	0.0	6.64E-01	0.0	7.40E-01	2.720	0.340
400940	0.0	0.0	0.0	0.0	0.0	1.13E-02	0.0	0.0	5.80E-01	0.0	0.0	0.0
370950	2.77E-01	0.0	0.0	0.0	0.0	3.16E-03	0.0	6.07E 00	1.41E 00	5.80E 00	3.420	0.0
380950	1.44E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.73E 00	0.0	2.500	0.0
390950	1.06E-03	0.0	0.0	0.0	0.0	3.94E-02	0.0	0.0	1.50E 00	0.0	2.020	0.210
400950	1.23E-07	0.020	0.0	0.0	0.0	5.71E-02	0.0	0.0	0.0	0.0	0.883	0.821
410951	2.14E-06	0.0	0.0	0.0	1.000	3.93E-01	0.0	0.0	0.0	0.0	0.235	1.000
410950	2.29E-07	0.0	0.0	0.0	0.0	3.93E-01	0.0	0.0	0.0	0.0	0.809	0.945
420950	0.0	0.0	0.0	0.0	0.0	1.41E-01	0.0	0.0	6.40E-02	0.0	0.0	0.0
390960	5.02E-03	0.0	0.0	0.0	0.0	1.65E-01	0.0	3.04E 00	5.90E 00	2.58E 00	5.020	0.700
400960	0.0	0.0	0.0	0.0	0.0	1.61E-02	0.0	3.04E 00	0.0	2.58E 00	0.0	0.0
410960	8.37E-06	0.0	0.0	0.0	0.0	1.79E 00	0.0	5.41E-04	7.00E-03	3.60E-03	2.610	0.900
420960	0.0	0.0	0.0	0.0	0.0	3.73E-02	0.0	0.0	0.0	0.0	0.0	0.0
390970	1.16E-01	0.0	0.0	0.0	0.0	4.92E-02	0.0	4.54E 00	5.85E 00	6.10E 00	2.250	0.0
400970	1.13E-05	0.960	0.0	0.0	0.0	8.27E-02	0.0	1.33E 00	0.0	1.23E 00	0.855	0.081
410971	1.16E-02	0.0	0.0	0.0	1.000	5.11E-01	0.0	0.0	0.0	0.0	0.747	1.000
410970	1.60E-04	0.0	0.0	0.0	0.0	5.11E-01	0.0	0.0	0.0	0.0	1.160	0.580
420970	0.0	0.0	0.0	0.0	0.0	1.61E-01	0.0	0.0	1.50E-01	0.0	0.0	0.0
400980	1.16E-02	1.000	0.0	0.0	0.0	2.28E-02	0.0	1.90E-01	6.00E 00	1.00E-01	0.550	0.0
410981	5.78E-03	0.0	0.0	0.0	0.0	1.85E 00	0.0	6.07E-02	0.0	1.00E-01	2.040	0.0
410980	2.27E-04	0.0	0.0	0.0	0.0	1.85E 00	0.0	2.85E 00	2.00E-01	2.84E 00	2.830	0.530
420980	0.0	0.0	0.0	0.0	0.0	4.32E-02	0.0	2.39E 00	0.0	2.84E 00	0.0	0.0
410990	4.81E-03	0.0	0.0	0.0	0.0	4.72E-01	0.0	5.98E 00	6.30E 00	6.10E 00	1.150	0.0
420990	2.87E-06	0.870	0.0	0.0	0.0	1.84E-01	0.0	0.0	0.0	0.0	0.758	0.290
430991	3.21E-05	0.0	0.0	0.0	1.000	2.04E-01	0.0	0.0	0.0	0.0	0.143	0.0
430990	1.04E-13	0.0	0.0	0.0	0.0	2.04E-01	0.0	0.0	0.0	0.0	0.114	0.0
440990	0.0	0.0	0.0	0.0	0.0	4.91E-01	0.0	0.0	0.0	0.0	0.0	0.0
411000	3.85E-03	0.0	0.0	0.0	0.0	1.61E 00	0.0	5.98E 00	6.40E 00	7.10E 00	2.990	0.180
421000	0.0	0.0	0.0	0.0	0.0	4.91E-02	0.0	0.0	0.0	0.0	0.0	0.0
431000	4.08E-02	0.0	0.0	0.0	0.0	7.66E-01	0.0	0.0	0.0	0.0	1.460	0.0
441000	0.0	0.0	0.0	0.0	0.0	1.26E-01	0.0	0.0	0.0	0.0	0.0	0.0
411010	1.16E-02	0.0	0.0	0.0	0.0	3.93E-01	0.0	2.37E 00	6.50E 00	2.95E 00	1.150	0.0
421010	7.91E-04	0.0	0.0	0.0	0.0	2.08E-01	0.0	2.37E 00	0.0	2.95E 00	2.070	0.800
431010	9.25E-04	0.0	0.0	0.0	0.0	1.98E-01	0.0	0.0	0.0	0.0	0.803	0.410
441010	0.0	0.0	0.0	0.0	0.0	5.11E-01	0.0	0.0	0.0	0.0	0.0	0.0
421020	1.05E-03	0.500	0.0	0.0	0.0	5.50E-02	0.0	3.98E 00	6.60E 00	5.99E 00	0.492	0.0
431021	2.57E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.980	0.720
431020	1.39E-01	0.0	0.0	0.0	0.0	7.27E-01	0.0	0.0	0.0	0.0	1.920	0.0
441020	0.0	0.0	0.0	0.0	0.0	1.28E-01	0.0	0.0	0.0	0.0	0.0	0.0
421030	1.12E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.60E 00	0.0	1.200	0.0
431030	1.39E-02	0.0	0.0	0.0	0.0	1.78E-01	0.0	1.47E 00	0.0	2.83E 00	1.060	0.0
441030	2.03E-07	1.000	0.0	0.0	0.0	4.24E-01	0.0	1.38E 00	0.0	2.83E 00	0.551	0.880
451031	2.03E-04	0.0	0.0	0.0	1.000	3.73E-01	0.0	0.0	0.0	0.0	0.040	0.0
451030	0.0	0.0	0.0	0.0	0.0	0.0	0.071	0.0	0.0	0.0	0.0	0.0
421040	8.89E-03	0.0	0.0	0.0	0.0	6.21E-02	0.0	1.71E 00	5.00E 00	5.93E 00	1.814	0.960
431040	6.42E-04	0.0	0.0	0.0	0.0	6.29E-01	0.0	0.0	0.0	0.0	4.520	0.770

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
441040	0.0	0.0	0.0	0.0	0.0	7.46E-02	0.0	0.0	0.0	0.0	0.0	0.0
451041	2.62E-03	0.0	0.0	0.0	1.000	1.37E 00	0.0	0.0	0.0	0.0	0.129	0.0
451040	1.61E-02	0.0	0.005	0.0	0.0	1.37E 00	0.0	0.0	0.0	0.0	1.007	0.012
461040	0.0	0.0	0.0	0.0	0.0	9.81E-02	0.0	0.0	0.0	0.0	0.0	0.0
421050	1.73E-02	0.0	0.0	0.0	0.0	2.63E-01	0.0	1.14E 00	3.30E 00	2.60E 00	2.200	0.0
431050	8.66E-02	0.0	0.0	0.0	0.0	1.52E-01	0.0	5.69E-01	0.0	1.30E 00	1.390	0.0
441050	4.34E-05	1.000	0.0	0.0	0.0	1.89E-01	0.0	0.0	0.0	0.0	1.140	0.640
451051	1.54E-02	0.0	0.0	0.0	1.000	3.45E-01	0.0	0.0	0.0	0.0	0.129	0.0
451050	5.36E-06	0.0	0.0	0.0	0.0	3.45E-01	0.715	0.0	0.0	0.0	0.227	0.350
461050	0.0	0.0	0.0	0.0	0.0	4.16E-01	0.0	0.0	0.0	0.0	0.0	0.0
431060	1.87E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.70E 00	0.0	2.500	0.200
441060	2.19E-08	0.0	0.0	0.0	0.0	2.55E-02	0.0	9.01E-01	0.0	4.57E 00	0.010	0.0
451061	8.89E-05	0.0	0.0	0.0	0.0	1.27E 00	0.0	0.0	0.0	0.0	3.180	0.910
451060	2.31E-02	0.0	0.0	0.0	0.0	1.27E 00	0.0	0.0	0.0	0.0	1.630	0.120
461060	0.0	0.0	0.0	0.0	0.0	1.05E-01	0.0	0.0	0.0	0.0	0.0	0.0
441070	2.75E-03	0.0	0.0	0.0	0.0	3.02E-02	0.0	7.58E-01	2.00E 00	3.60E 00	1.490	0.134
451070	5.25E-04	0.200	0.0	0.0	0.0	3.14E-01	0.0	0.0	0.0	0.0	0.796	0.390
461071	3.15E-02	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.210	1.000
461070	0.0	0.0	0.0	0.0	0.0	4.24E-01	0.0	0.0	0.0	0.0	0.014	0.0
471070	0.0	0.0	0.0	0.0	0.0	3.28E-01	0.010	0.0	0.0	0.0	0.0	0.0
441080	2.57E-03	0.0	0.0	0.0	0.0	3.93E-03	0.0	3.04E-01	6.00E-01	2.10E 00	0.501	0.0
451080	4.08E-02	0.0	0.0	0.0	0.0	1.15E 00	0.0	0.0	0.0	0.0	2.150	0.160
461080	0.0	0.0	0.0	0.0	0.0	1.06E-01	0.016	0.0	0.0	0.0	0.0	0.0
471080	4.77E-03	0.0	0.025	0.0	0.0	1.51E 00	0.0	0.0	0.0	0.0	0.623	0.018
481080	0.0	0.0	0.0	0.0	0.0	1.26E-01	0.0	0.0	0.0	0.0	0.0	0.0
451090	2.31E-02	0.0	0.0	0.0	0.0	2.85E-01	0.0	1.06E-01	3.20E-01	2.80E 00	1.550	0.530
461091	2.46E-03	0.0	0.0	0.0	1.000	3.92E-01	0.0	0.0	0.0	0.0	0.188	0.0
461090	1.43E-05	1.000	0.0	0.0	0.0	3.92E-01	0.0	0.0	0.0	0.0	0.413	0.0
471091	1.73E-02	0.0	0.0	0.0	1.000	4.32E-01	0.0	0.0	0.0	0.0	0.088	0.0
471090	0.0	0.0	0.0	0.0	0.0	4.32E-01	0.033	0.0	0.0	0.0	0.0	0.0
481090	1.77E-08	0.0	1.000	1.000	0.0	0.0	0.0	0.0	0.0	0.0	0.080	0.0
451100	1.39E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.50E-01	0.0	2.320	0.024
461100	0.0	0.0	0.0	0.0	0.0	8.63E-02	0.167	7.59E-02	0.0	0.0	0.0	0.0
471101	3.17E-08	0.0	0.0	0.0	0.130	1.75E 00	0.0	0.0	0.0	0.0	2.720	0.960
471100	2.84E-02	0.0	0.0	0.0	0.0	1.75E 00	0.0	0.0	0.0	0.0	1.224	0.027
481100	0.0	0.0	0.0	0.0	0.0	1.34E-01	0.032	7.59E-07	0.0	0.0	0.0	0.0
461111	3.50E-05	1.000	0.0	0.0	0.750	2.65E-01	0.0	3.79E-03	7.60E-04	2.00E-02	0.373	0.0
461110	5.25E-04	1.000	0.0	0.0	0.0	2.65E-01	0.0	6.83E-02	7.60E-02	4.40E-01	0.848	0.0
471111	9.37E-03	0.0	0.0	0.0	1.000	4.51E-01	0.0	0.0	0.0	0.0	0.070	0.0
471110	1.07E-06	0.0	0.0	0.0	0.0	4.51E-01	0.0	0.0	0.0	0.0	0.404	0.0
481111	2.38E-04	0.0	0.0	0.0	1.000	5.50E-01	0.0	0.0	0.0	0.0	0.396	0.624
481110	0.0	0.0	0.0	0.0	0.0	5.50E-01	0.0	0.0	0.0	0.0	0.0	0.0
461120	9.17E-06	0.0	0.0	0.0	0.0	4.71E-02	0.0	4.17E-02	4.60E-02	2.40E-01	0.096	0.0
471120	6.02E-05	0.0	0.0	0.0	0.0	1.77E 00	0.0	0.0	0.0	0.0	2.315	0.422
481120	0.0	0.0	0.0	0.0	0.0	1.37E-01	1.000	0.0	0.0	0.0	0.0	0.0
461130	8.25E-03	0.100	0.0	0.0	0.0	1.18E-01	0.0	4.17E-02	3.45E-02	2.00E-02	1.250	0.0
471131	9.63E-03	0.0	0.0	0.0	0.0	4.32E-01	0.0	0.0	0.0	0.0	0.830	0.0
471130	3.63E-05	0.0	0.0	0.0	0.0	4.32E-01	0.0	0.0	0.0	0.0	1.350	0.740
481131	1.57E-09	0.0	0.0	0.0	0.001	4.56E-01	0.0	0.0	0.0	0.0	0.223	0.001
481130	0.0	0.0	0.0	0.0	0.0	4.56E-01	0.0	0.0	0.0	0.0	0.0	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
491130	0.0	0.0	0.0	0.0	0.0	4.83E-01	0.670	0.0	0.0	0.0	0.0	0.0
461140	4.81E-03	0.0	0.0	0.0	0.0	1.45E-02	0.0	3.79E-02	4.00E-02	2.00E-02	1.250	0.0
471140	1.39E-01	0.0	0.0	0.0	0.0	1.59E 00	0.0	0.0	0.0	0.0	2.280	0.250
481140	0.0	0.0	0.0	0.0	0.0	7.86E-02	0.113	0.0	0.0	0.0	0.0	0.0
491141	1.60E-07	0.0	0.035	0.0	0.965	1.55E 00	0.0	0.0	0.0	0.0	0.240	0.240
491140	9.63E-03	0.0	0.019	0.0	0.0	1.55E 00	0.0	0.0	0.0	0.0	0.794	0.001
501140	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
461150	1.54E-02	0.280	0.0	0.0	0.0	4.32E-02	0.0	3.79E-02	3.70E-02	7.60E-03	2.000	0.0
471151	3.47E-02	0.0	0.0	0.0	0.0	3.83E-01	0.0	0.0	0.0	0.0	1.360	0.0
471150	5.78E-04	0.090	0.0	0.0	0.0	3.83E-01	0.0	1.90E-03	0.0	6.00E-04	1.990	0.570
481151	1.87E-07	0.0	0.0	0.0	0.0	2.02E-01	0.0	0.0	0.0	0.0	0.687	0.147
481150	3.60E-06	1.000	0.0	0.0	0.0	2.02E-01	0.0	0.0	0.0	0.0	0.562	0.344
491151	4.28E-05	0.0	0.0	0.0	0.950	3.36E-01	0.0	0.0	0.0	0.0	0.335	0.951
491150	0.0	0.0	0.0	0.0	0.0	3.36E-01	0.780	0.0	0.0	0.0	0.198	0.0
501150	0.0	0.0	0.0	0.0	0.0	2.59E-01	0.0	0.0	0.0	0.0	0.0	0.0
471160	4.62E-03	0.0	0.0	0.0	0.0	1.39E 00	0.0	3.79E-02	3.80E-02	0.0	2.700	0.0
481160	0.0	0.0	0.0	0.0	0.0	2.75E-02	0.0	0.0	0.0	0.0	0.0	0.0
491161	2.14E-04	0.0	0.0	0.0	0.0	1.23E 00	0.0	0.0	0.0	0.0	2.830	0.896
491160	4.95E-02	0.0	0.0	0.0	0.0	1.23E 00	0.0	0.0	0.0	0.0	1.400	0.009
501160	0.0	0.0	0.0	0.0	0.0	4.32E-02	1.000	1.14E-02	0.0	0.0	0.0	0.0
471170	1.05E-02	0.0	0.0	0.0	0.0	3.30E-01	0.0	4.17E-02	4.00E-02	2.20E-02	1.250	0.0
481171	5.66E-05	0.440	0.0	0.0	0.0	6.09E-02	0.0	0.0	0.0	0.0	1.660	0.570
481170	8.02E-05	1.000	0.0	0.0	0.0	6.09E-02	0.0	0.0	0.0	0.0	1.780	0.720
491171	9.98E-05	0.0	0.0	0.0	0.470	2.96E-01	0.0	0.0	0.0	0.0	0.533	0.280
491170	2.63E-04	0.0	0.0	0.0	0.0	2.96E-01	0.0	0.0	0.0	0.0	0.976	0.570
501171	5.73E-07	0.0	0.0	0.0	1.000	1.65E-01	0.0	0.0	0.0	0.0	0.317	0.0
501170	0.0	0.0	0.0	0.0	0.0	1.65E-01	0.0	0.0	0.0	0.0	0.0	0.0
481180	2.36E-04	0.0	0.0	0.0	0.0	8.45E-03	0.0	3.82E-02	4.00E-02	2.00E-03	0.266	0.0
491181	2.63E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.280	0.810
491180	1.39E-01	0.0	0.0	0.0	0.0	1.12E 00	0.0	0.0	0.0	0.0	1.920	0.096
501180	0.0	0.0	0.0	0.0	0.0	2.63E-02	1.000	0.0	0.0	0.0	0.0	0.0
481191	4.28E-03	1.000	0.0	0.0	0.0	3.06E-02	0.0	3.82E-02	2.00E-02	8.00E-03	1.500	0.0
481190	1.16E-03	1.000	0.0	0.0	0.0	3.06E-02	0.0	0.0	2.00E-02	0.0	1.500	0.0
491191	6.42E-04	0.0	0.0	0.0	0.050	2.73E-01	0.0	0.0	0.0	0.0	1.120	0.011
491190	5.50E-03	0.050	0.0	0.0	0.0	2.73E-01	0.0	0.0	0.0	0.0	1.470	0.560
501191	3.21E-08	0.0	0.0	0.0	1.000	1.00E-01	0.0	0.0	0.0	0.0	0.089	0.0
501190	0.0	0.0	0.0	0.0	0.0	1.00E-01	0.0	0.0	0.0	0.0	0.0	0.0
481200	1.16E-02	0.500	0.0	0.0	0.0	0.0	0.0	0.0	4.10E-02	0.0	1.250	0.0
491201	2.17E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.550	0.069
491200	1.51E-02	0.0	0.0	0.0	0.0	1.05E 00	0.0	1.91E-02	0.0	9.63E-04	3.610	0.670
501200	0.0	0.0	0.0	0.0	0.0	1.57E-02	0.007	1.91E-02	0.0	9.63E-04	0.0	0.0
481210	3.30E-03	1.000	0.0	0.0	0.0	1.53E-02	0.0	9.54E-04	4.20E-02	4.82E-03	1.980	0.750
491211	3.73E-03	0.0	0.0	0.0	0.0	2.57E-01	0.0	9.54E-04	0.0	4.82E-03	1.530	0.0
491210	2.31E-02	0.0	0.0	0.0	0.0	2.57E-01	0.0	1.91E-03	0.0	1.02E-02	2.060	0.450
501211	2.89E-10	0.0	0.0	0.0	0.0	5.69E-02	0.0	1.91E-03	0.0	9.63E-03	0.177	0.0
501210	7.13E-06	0.0	0.0	0.0	0.0	5.69E-02	0.0	5.34E-02	0.0	5.78E-02	0.112	0.0
511210	0.0	0.0	0.0	0.0	0.0	1.88E-01	0.010	0.0	0.0	0.0	0.0	0.0
491220	8.66E-02	0.0	0.0	0.0	0.0	9.93E-01	0.0	2.43E-02	4.50E-02	9.63E-04	4.120	0.520
501220	0.0	0.0	0.0	0.0	0.0	9.03E-03	0.005	2.48E-02	0.0	9.63E-04	0.0	0.0
511221	2.75E-03	0.0	0.0	0.0	1.000	7.26E-01	0.0	0.0	0.0	0.0	0.162	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
511220	2.87E-06	0.0	0.031	0.0	0.0	7.26E-01	0.0	0.0	0.0	0.0	1.070	0.430
521220	0.0	0.0	0.0	0.0	0.0	1.02E-01	0.333	0.0	0.0	0.0	0.0	0.0
491231	1.93E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.50E-02	0.0	3.200	0.710
491230	6.93E-02	1.000	0.0	0.0	0.0	2.45E-01	0.0	2.29E-03	5.00E-04	9.63E-04	3.200	0.710
501231	6.42E-08	0.0	0.0	0.0	0.0	3.30E-02	0.0	2.29E-03	0.0	9.63E-04	0.575	0.0
501230	2.89E-04	0.0	0.0	0.0	0.0	3.30E-02	0.0	5.34E-02	0.0	0.0	0.682	0.0
511230	0.0	0.0	0.0	0.0	0.0	1.71E-01	0.014	0.0	0.0	0.0	0.0	0.0
521231	6.86E-08	0.0	0.0	0.0	1.000	3.30E-01	0.0	0.0	0.0	0.0	0.248	0.0
521230	0.0	0.0	0.0	0.0	0.0	3.30E-01	0.0	0.0	0.0	0.0	0.0	0.0
491240	1.73E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.50E-02	0.0	4.430	0.474
501240	0.0	0.0	0.0	0.0	0.0	5.38E-03	0.962	7.63E-02	0.0	0.0	0.0	0.0
511241	7.45E-03	0.0	0.0	0.0	0.800	6.52E-01	0.0	0.0	0.0	0.0	0.440	0.790
511240	1.34E-07	0.0	0.0	0.0	0.0	6.52E-01	0.0	0.0	0.0	0.0	2.280	0.820
521240	0.0	0.0	0.0	0.0	0.0	6.48E-02	0.714	0.0	0.0	0.0	0.0	0.0
501251	1.19E-03	0.0	0.0	0.0	0.0	2.02E-02	0.0	4.20E-02	2.50E-02	1.93E-03	1.187	0.298
501250	8.53E-07	0.0	0.0	0.0	0.0	2.02E-02	0.0	4.58E-02	4.00E-02	1.37E-01	1.021	0.091
511250	8.14E-09	0.390	0.0	0.0	0.0	1.57E-01	0.0	0.0	0.0	0.0	0.566	0.550
521251	1.38E-07	0.0	0.0	0.0	1.000	1.96E-01	0.0	0.0	0.0	0.0	0.145	0.0
521250	0.0	0.0	0.0	0.0	0.0	1.96E-01	0.0	0.0	0.0	0.0	0.0	0.0
501260	2.20E-13	1.000	0.0	0.0	0.0	3.30E-03	0.0	2.39E-01	4.00E-02	3.85E-01	0.090	0.0
511261	6.08E-04	0.0	0.0	0.0	0.990	6.08E-01	0.0	0.0	4.00E-02	0.0	0.098	0.380
511260	6.42E-07	0.0	0.0	0.0	0.0	6.08E-01	0.0	0.0	0.0	0.0	1.850	0.580
521260	0.0	0.0	0.0	0.0	0.0	3.53E-02	0.100	0.0	0.0	0.0	0.0	0.0
501271	2.89E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.630	0.310
501270	9.17E-05	0.0	0.0	0.0	0.0	1.26E-02	0.0	5.73E-01	1.20E-01	7.51E-01	1.810	0.530
511270	2.07E-06	0.220	0.0	0.0	0.0	1.47E-01	0.0	2.39E-02	0.0	1.93E-02	0.927	0.507
521271	7.36E-08	0.0	0.0	0.0	0.985	9.74E-02	0.0	0.0	0.0	0.0	0.093	0.0
521270	2.05E-05	0.0	0.0	0.0	0.0	9.74E-02	0.0	0.0	0.0	0.0	0.275	0.015
531270	0.0	0.0	0.0	0.0	0.0	2.95E-01	0.0	0.0	0.0	0.0	0.0	0.0
501280	1.96E-04	0.970	0.0	0.0	0.0	2.12E-03	0.0	8.83E-01	3.50E-01	7.70E-01	0.846	0.650
511281	1.05E-03	0.0	0.0	0.0	0.0	5.69E-01	0.0	1.19E-01	0.0	0.0	2.763	0.597
511280	2.14E-05	0.0	0.0	0.0	0.0	5.69E-01	0.0	1.91E-01	3.50E-02	1.93E-01	3.557	0.858
521280	0.0	0.0	0.0	0.0	0.0	1.57E-02	0.108	0.0	0.0	0.0	0.0	0.0
531280	4.62E-04	0.0	0.063	0.0	0.0	1.09E 00	0.0	7.16E-05	3.00E-05	3.85E-04	0.959	0.173
541280	0.0	0.0	0.0	0.0	0.0	4.91E-02	0.0	0.0	0.0	0.0	0.0	0.0
501291	1.93E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.30E 00	0.0	1.930	0.570
501290	1.28E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.930	0.570
511290	4.48E-05	0.360	0.0	0.0	0.0	1.37E-01	0.0	1.91E 00	0.0	1.93E 00	1.360	0.440
521291	2.36E-07	0.0	0.0	0.0	0.640	3.93E-02	0.0	0.0	0.0	0.0	0.334	0.120
521290	1.67E-04	0.0	0.0	0.0	0.0	3.93E-02	0.0	0.0	0.0	0.0	0.612	0.150
531290	1.29E-15	0.0	0.0	0.0	0.0	2.43E-01	0.0	0.0	0.0	0.0	0.073	0.0
541291	1.00E-06	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.236	0.0
541290	0.0	0.0	0.0	0.0	0.0	1.77E-01	0.0	0.0	4.00E-04	0.0	0.0	0.0
501300	4.44E-03	0.500	0.0	0.0	0.0	1.37E-03	0.0	1.91E 00	2.00E 00	1.93E 00	0.0	0.0
511301	1.65E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.320	0.630
511300	3.50E-04	0.0	0.0	0.0	0.0	5.40E-01	0.0	0.0	0.0	0.0	3.320	0.630
521300	0.0	0.0	0.0	0.0	0.0	5.89E-03	0.130	0.0	0.0	0.0	0.0	0.0
531301	1.26E-03	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.100	0.0
531300	1.55E-05	0.0	0.0	0.0	0.0	8.84E-01	0.0	4.77E-04	5.00E-04	5.01E-03	1.480	0.780
541300	0.0	0.0	0.0	0.0	0.0	2.99E-02	0.0	0.0	0.0	0.0	0.0	0.0

NUCLIDE LIBRARY FOR A1 REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
501310	3.40E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.20E 00	0.0	2.000	0.0
511310	4.62E-04	0.150	0.0	0.0	0.0	1.30E-01	0.0	2.58E 00	0.0	2.99E 00	1.850	0.510
521311	6.42E-06	0.0	0.0	0.0	0.180	1.49E-02	0.0	3.82E-02	0.0	1.25E-01	1.638	0.805
521310	4.62E-04	0.0	0.0	0.0	0.0	1.49E-02	0.0	1.53E-01	0.0	5.20E-01	1.150	0.250
531310	9.97E-07	0.008	0.0	0.0	0.0	1.96E-01	0.0	1.91E-01	0.0	0.0	0.694	0.574
541311	6.80E-07	0.0	0.0	0.0	1.000	1.51E-01	0.0	0.0	0.0	9.63E-03	0.164	0.0
541310	0.0	0.0	0.0	0.0	0.0	1.51E-01	0.0	0.0	0.0	0.0	0.0	0.0
501320	5.25E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.56E 00	0.0	2.000	0.0
511321	1.54E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.360	0.0
511320	5.50E-03	0.0	0.0	0.0	0.0	5.10E-01	0.0	3.25E 00	0.0	4.91E 00	3.360	0.0
521320	2.47E-06	0.0	0.0	0.0	0.0	2.36E-03	0.0	9.54E-01	0.0	1.73E-01	0.230	0.380
531320	8.37E-05	0.0	0.0	0.0	0.0	7.15E-01	0.0	0.0	1.40E-01	0.0	2.700	0.780
541320	0.0	0.0	0.0	0.0	0.0	3.34E-02	0.962	0.0	0.0	0.0	0.0	0.0
511330	2.75E-03	0.720	0.0	0.0	0.0	1.22E-01	0.0	3.53E 00	3.22E 00	2.89E 00	2.760	0.0
521331	2.31E-04	0.0	0.0	0.0	0.130	6.28E-03	0.0	1.05E 00	1.86E 00	9.63E-01	2.350	0.610
521330	9.24E-04	0.0	0.0	0.0	0.0	6.28E-03	0.0	1.05E 00	0.0	9.63E-01	1.920	0.680
531330	9.17E-06	0.024	0.0	0.0	0.0	1.53E-01	0.0	4.77E-01	4.20E-01	1.93E-01	1.100	0.510
541331	3.55E-06	0.0	0.0	0.0	1.000	2.08E-01	0.0	0.0	0.0	0.0	0.233	0.0
541330	1.52E-06	0.0	0.0	0.0	0.0	2.08E-01	0.0	0.0	0.0	1.64E 00	0.182	0.0
551330	0.0	0.0	0.0	0.0	0.0	1.97E-01	0.085	0.0	0.0	0.0	0.0	0.0
511340	4.62E-01	0.0	0.0	0.0	0.0	4.71E-01	0.0	2.86E 00	5.65E 00	3.37E 00	3.680	0.0
521340	2.75E-04	0.0	0.0	0.0	0.0	1.10E-03	0.0	3.53E 00	0.0	3.85E 00	2.800	0.0
531340	2.19E-04	0.0	0.0	0.0	0.0	5.73E-01	0.0	8.59E-01	7.40E-01	0.0	3.270	0.814
541340	0.0	0.0	0.0	0.0	0.0	6.21E-02	0.962	0.0	2.10E-01	0.0	0.0	0.0
551341	6.64E-05	0.0	0.0	0.0	0.990	6.49E-01	0.0	0.0	0.0	0.0	0.153	0.098
551340	1.07E-08	0.0	0.0	0.0	0.0	6.49E-01	0.0	0.0	0.0	0.0	1.770	0.890
561340	0.0	0.0	0.0	0.0	0.0	1.02E-02	0.038	0.0	0.0	0.0	0.0	0.0
521350	2.39E-02	0.0	0.0	0.0	0.0	3.14E-03	0.0	4.10E 00	4.92E 00	1.06E 00	2.630	0.0
531350	2.87E-05	0.300	0.0	0.0	0.0	1.19E-01	0.0	1.81E 00	0.0	5.49E 00	2.280	0.840
541351	7.41E-04	0.0	0.0	0.0	1.000	9.82E-01	0.0	0.0	0.0	0.0	0.527	1.000
541350	2.09E-05	0.0	0.0	0.0	0.0	4.24E 00	0.0	2.86E-01	1.08E 00	4.53E-01	0.567	0.460
551351	2.18E-04	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	1.621	1.000
551350	7.32E-15	0.0	0.0	0.0	0.0	1.14E-01	0.0	0.0	0.0	0.0	0.082	0.0
561351	6.71E-06	0.0	0.0	0.0	1.000	3.93E-02	0.0	0.0	0.0	0.0	0.268	1.000
561350	0.0	0.0	0.0	0.0	0.0	3.93E-02	0.0	0.0	0.0	0.0	0.0	0.0
531360	8.35E-03	0.0	0.0	0.0	0.0	4.61E-01	0.0	2.96E 00	2.63E 00	2.02E 00	4.250	0.520
541360	0.0	0.0	0.0	0.0	0.0	1.67E-01	0.0	3.21E 00	3.07E 00	4.35E 00	0.0	0.0
551360	6.17E-07	0.0	0.0	0.0	0.0	2.97E-01	0.0	6.49E-03	1.00E-01	1.06E-01	2.610	0.940
561360	0.0	0.0	0.0	0.0	0.0	6.10E-03	0.010	0.0	0.0	0.0	0.0	0.0
531370	3.01E-02	0.0	0.0	0.0	0.0	9.43E-02	0.0	4.68E 00	6.05E 00	4.91E 00	0.176	0.0
541370	2.96E-03	0.0	0.0	0.0	0.0	5.11E-01	0.0	1.24E 00	0.0	1.47E 00	2.060	0.150
551370	7.32E-10	0.935	0.0	0.0	0.0	3.93E-02	0.0	0.0	1.50E-01	0.0	0.273	0.0
561371	4.53E-03	0.0	0.0	0.0	1.000	2.16E-02	0.0	0.0	0.0	0.0	0.662	1.000
561370	0.0	0.0	0.0	0.0	0.0	2.16E-02	0.0	0.0	0.0	0.0	0.0	0.0
531380	1.17E-01	0.0	0.0	0.0	0.0	3.77E-01	0.0	3.25E 00	6.00E 00	3.56E 00	3.420	0.0
541380	6.80E-04	0.0	0.0	0.0	0.0	7.86E-02	0.0	2.00E 00	0.0	2.13E 00	1.420	0.300
551380	3.59E-04	0.0	0.0	0.0	0.0	1.04E-01	0.0	2.86E-01	0.0	3.85E-01	3.620	0.690
561380	0.0	0.0	0.0	0.0	0.0	3.15E-03	0.0	0.0	0.0	0.0	0.0	0.0
531390	3.47E-01	0.0	0.0	0.0	0.0	7.46E-02	0.0	1.72E 00	4.80E 00	1.64E 00	2.000	0.0
541390	1.61E-02	0.0	0.0	0.0	0.0	3.38E-01	0.0	2.77E 00	0.0	2.70E 00	2.000	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
551390	1.22E-03	0.0	0.0	0.0	0.0	2.00E-02	0.0	1.15E 00	9.60E-01	1.18E 00	2.390	0.530
561390	1.39E-04	0.0	0.0	0.0	0.0	9.44E-03	0.0	9.54E-02	7.00E-02	1.35E-01	0.936	0.007
571390	0.0	0.0	0.0	0.0	0.0	1.81E-02	0.0	0.0	0.0	0.0	0.0	0.0
541400	4.33E-02	0.0	0.0	0.0	0.0	6.29E-02	0.0	3.53E 00	3.41E 00	3.18E 00	1.690	0.0
551400	1.05E-02	0.0	0.0	0.0	0.0	7.19E-02	0.0	2.20E 00	1.97E 00	2.02E 00	3.490	0.460
561400	6.27E-07	0.0	0.0	0.0	0.0	1.18E-03	0.0	2.86E-01	3.10E-01	0.0	0.569	0.470
571400	4.79E-06	0.0	0.0	0.0	0.0	9.64E-02	0.0	0.0	0.0	1.93E-01	2.800	0.770
581400	0.0	0.0	0.0	0.0	0.0	1.58E-02	0.0	0.0	8.10E-02	0.0	0.0	0.0
541410	3.47E-01	0.0	0.0	0.0	0.0	3.10E-01	0.0	1.72E 00	1.23E 00	1.06E 00	2.670	0.0
551410	2.89E-02	0.0	0.0	0.0	0.0	1.65E-02	0.0	2.77E 00	3.01E 00	3.56E 00	2.040	0.0
561410	6.42E-04	0.0	0.0	0.0	0.0	3.74E-03	0.0	1.15E 00	1.57E 00	8.67E-01	1.370	0.0
571410	4.94E-05	0.0	0.0	0.0	0.0	2.95E-02	0.0	9.54E-02	9.20E-02	0.0	1.010	0.027
581410	2.48E-07	0.0	0.0	0.0	0.0	9.45E-02	0.0	0.0	0.0	0.0	0.332	0.0
591410	0.0	0.0	0.0	0.0	0.0	4.34E-02	0.0	0.0	0.0	0.0	0.0	0.0
541420	4.62E-01	0.0	0.0	0.0	0.0	6.56E-02	0.0	3.34E-01	3.30E-01	2.99E-01	3.000	0.0
551420	3.01E-01	0.0	0.0	0.0	0.0	6.49E-02	0.0	2.91E 00	2.94E 00	2.76E 00	3.000	0.0
561420	1.05E-03	0.0	0.0	0.0	0.0	3.93E-04	0.0	2.10E 00	2.13E 00	1.76E 00	0.927	0.0
571420	1.26E-04	0.0	0.0	0.0	0.0	1.67E-01	0.0	2.86E-01	2.90E-01	0.0	3.309	0.715
581420	0.0	0.0	0.0	0.0	0.0	1.42E-01	0.0	0.0	0.0	0.0	0.0	0.0
591420	1.00E-05	0.0	0.0	0.0	0.0	2.48E-01	0.0	0.0	0.0	0.0	2.420	0.648
601420	0.0	0.0	0.0	0.0	0.0	5.92E-02	0.0	0.0	0.0	0.0	0.0	0.0
541430	6.93E-01	0.0	0.0	0.0	0.0	3.63E-01	0.0	2.96E 00	5.20E-02	2.55E 00	3.090	0.0
551430	3.47E-01	0.0	0.0	0.0	0.0	1.57E-02	0.0	2.96E 00	1.35E 00	2.55E 00	2.870	0.140
561430	5.78E-02	0.0	0.0	0.0	0.0	1.57E-03	0.0	0.0	2.50E 00	0.0	1.870	0.0
571430	8.25E-04	0.0	0.0	0.0	0.0	5.59E-02	0.0	0.0	1.20E 00	0.0	1.560	0.190
581430	5.83E-06	0.0	0.0	0.0	0.0	2.35E-01	0.0	0.0	0.0	0.0	0.766	0.396
591430	5.86E-07	0.0	0.0	0.0	0.0	1.06E-01	0.0	0.0	0.0	0.0	0.366	0.0
601430	0.0	0.0	0.0	0.0	0.0	3.47E-01	0.0	0.0	0.0	0.0	0.0	0.0
571440	1.73E-01	0.0	0.0	0.0	0.0	3.38E-01	0.0	5.54E 00	4.50E 00	3.08E 00	2.610	0.0
581440	2.82E-08	0.0	0.0	0.0	0.0	7.88E-02	0.0	2.86E-01	0.0	5.68E-01	0.148	0.270
591440	6.68E-04	0.0	0.0	0.0	0.0	8.28E-01	0.0	0.0	0.0	1.35E-01	1.255	0.024
601440	0.0	0.0	0.0	0.0	0.0	8.36E-02	0.0	0.0	0.0	0.0	0.0	0.0
581450	3.85E-03	0.0	0.0	0.0	0.0	4.10E-01	0.0	4.01E 00	4.80E 00	3.01E 00	1.070	0.0
591450	3.22E-05	0.0	0.0	0.0	0.0	2.96E-01	0.0	0.0	0.0	0.0	0.730	0.050
601450	0.0	0.0	0.0	0.0	0.0	4.93E-01	0.0	0.0	0.0	0.0	0.0	0.0
581460	8.25E-04	0.0	0.0	0.0	0.0	1.26E-01	0.0	3.05E 00	4.20E 00	2.50E 00	0.474	0.450
591460	4.81E-04	0.0	0.0	0.0	0.0	1.68E 00	0.0	9.54E-02	0.0	0.0	2.710	0.620
601460	0.0	0.0	0.0	0.0	0.0	1.16E-01	0.0	0.0	0.0	0.0	0.0	0.0
581470	1.07E-02	0.0	0.0	0.0	0.0	5.91E-01	0.0	6.20E-01	3.50E 00	5.30E-01	2.000	0.0
591470	9.63E-04	0.0	0.0	0.0	0.0	5.81E-01	0.0	6.20E-01	0.0	5.30E-01	1.060	0.210
601470	7.23E-07	0.0	0.0	0.0	0.0	6.51E-01	0.0	1.24E 00	0.0	1.06E 00	0.472	0.043
611470	8.38E-09	0.0	0.0	0.0	0.0	7.90E-01	0.522	0.0	0.0	0.0	0.087	0.0
621470	0.0	0.0	0.0	0.0	0.0	4.55E-01	0.0	0.0	0.0	0.0	0.0	0.0
581480	1.61E-02	0.0	0.0	0.0	0.0	1.69E-01	0.0	1.63E 00	2.50E 00	1.67E 00	2.000	0.0
591480	5.78E-03	0.0	0.0	0.0	0.0	3.15E 00	0.0	0.0	0.0	0.0	2.080	0.140
601480	0.0	0.0	0.0	0.0	0.0	1.48E-01	0.0	0.0	0.0	0.0	0.0	0.0
611481	1.91E-07	0.0	0.0	0.0	0.070	4.94E 00	0.0	0.0	0.0	0.0	2.142	0.910
611480	1.49E-06	0.0	0.0	0.0	0.0	4.94E 00	0.0	0.0	0.0	0.0	1.380	0.460
621480	0.0	0.0	0.0	0.0	0.0	1.11E-01	0.0	0.0	0.0	0.0	0.0	0.0
591490	5.02E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.80E 00	0.0	2.000	0.0

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
601490	1.07E-04	0.0	0.0	0.0	0.0	7.89E-01	0.0	1.24E 00	0.0	1.27E 00	0.866	0.390
611490	3.63E-06	0.0	0.0	0.0	0.0	1.38E 00	0.0	0.0	0.0	0.0	0.421	0.020
621490	0.0	0.0	0.0	0.0	0.0	6.32E-01	0.0	0.0	0.0	0.0	0.0	0.0
601500	0.0	0.0	0.0	0.0	0.0	1.64E-01	0.0	7.06E-01	1.50E 00	9.73E-01	0.0	0.0
611500	7.13E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.270	0.660
621500	0.0	0.0	0.0	0.0	0.0	1.46E-01	0.0	0.0	0.0	0.0	0.0	0.0
601510	9.63E-04	0.0	0.0	0.0	0.0	7.69E-01	0.0	4.58E-01	1.20E 00	7.70E-01	1.220	0.0
611510	6.88E-06	0.0	0.0	0.0	0.0	1.18E 00	0.0	1.91E-02	0.0	0.0	0.614	0.430
621510	2.52E-10	0.0	0.0	0.0	0.0	7.71E-01	0.0	0.0	0.0	0.0	0.294	0.0
631510	0.0	0.0	0.0	0.0	0.0	1.42E 00	0.322	0.0	0.0	0.0	0.0	0.0
611520	1.93E-03	0.0	0.0	0.0	0.0	4.74E 00	0.0	1.43E-01	8.50E-01	2.99E-01	2.260	0.550
621520	0.0	0.0	0.0	0.0	0.0	1.62E-01	0.0	1.43E-01	0.0	2.99E-01	0.0	0.0
631521	2.07E-05	0.0	0.230	0.0	0.0	5.38E 00	0.001	0.0	0.0	0.0	1.190	0.0
631520	1.83E-09	0.0	0.720	0.0	0.0	5.38E 00	0.0	0.0	0.0	0.0	2.060	0.0
641520	0.0	0.0	0.0	0.0	0.0	1.27E-01	0.0	0.0	0.0	0.0	0.0	0.0
611530	2.10E-03	0.0	0.0	0.0	0.0	1.01E 00	0.0	7.16E-02	3.60E-01	1.78E-01	0.825	0.0
621530	4.10E-06	0.0	0.0	0.0	0.0	7.90E-01	0.0	7.16E-02	4.70E-02	1.78E-01	0.373	0.005
631530	0.0	0.0	0.0	0.0	0.0	1.06E 00	0.0	0.0	0.0	0.0	0.0	0.0
641530	3.32E-08	0.0	1.000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.243	0.0
611540	4.62E-03	0.0	0.0	0.0	0.0	3.95E 00	0.0	4.29E-02	2.50E-01	1.40E-01	1.010	0.0
621540	0.0	0.0	0.0	0.0	0.0	1.50E-01	0.0	4.29E-02	0.0	1.40E-01	0.0	0.0
631540	1.37E-09	0.0	0.0	0.0	0.0	4.04E 00	0.0	0.0	0.0	0.0	1.580	0.840
641540	0.0	0.0	0.0	0.0	0.0	1.74E-01	0.0	0.0	0.0	0.0	0.0	0.0
621550	5.02E-04	0.0	0.0	0.0	0.0	6.64E-01	0.0	5.92E-02	1.30E-01	4.43E-01	0.680	0.180
631550	1.21E-08	0.0	0.0	0.0	0.0	8.11E-01	0.0	0.0	0.0	0.0	0.142	0.0
641550	0.0	0.0	0.0	0.0	0.0	8.91E-01	0.0	0.0	0.0	0.0	0.0	0.0
621560	2.05E-05	0.0	0.0	0.0	0.0	1.19E-01	0.0	2.48E-02	7.10E-02	2.12E-01	0.321	0.430
631560	5.35E-07	0.0	0.0	0.0	0.0	3.13E 00	0.0	0.0	0.0	0.0	1.780	0.760
641560	0.0	0.0	0.0	0.0	0.0	2.16E-01	0.0	0.0	0.0	0.0	0.0	0.0
621570	2.31E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.50E-02	0.0	0.921	0.620
631570	1.27E-05	0.0	0.0	0.0	0.0	6.33E-01	0.0	1.41E-02	0.0	1.43E-01	0.728	0.440
641570	0.0	0.0	0.0	0.0	0.0	1.04E 00	0.0	0.0	0.0	0.0	0.0	0.0
631580	2.51E-04	0.0	0.0	0.0	0.0	2.49E 00	0.0	1.91E-02	1.30E-02	3.85E-01	2.120	0.560
641580	0.0	0.0	0.0	0.0	0.0	2.38E-01	0.0	0.0	0.0	0.0	0.0	0.0
631590	6.42E-04	0.0	0.0	0.0	0.0	5.14E-01	0.0	5.25E-03	8.40E-03	1.01E-01	1.350	0.390
641590	1.07E-05	0.0	0.0	0.0	0.0	1.01E 00	0.0	5.25E-03	0.0	1.01E-01	0.410	0.120
651590	0.0	0.0	0.0	0.0	0.0	8.73E-01	0.0	0.0	0.0	0.0	0.0	0.0
631600	4.62E-03	0.0	0.0	0.0	0.0	2.02E 00	0.0	1.29E-02	3.90E-03	7.80E-03	1.490	0.0
641600	0.0	0.0	0.0	0.0	0.0	1.92E-01	0.0	1.29E-02	0.0	7.80E-03	0.0	0.0
651600	1.11E-07	0.0	0.0	0.0	0.0	3.51E 00	0.0	0.0	0.0	0.0	1.420	0.820
661600	0.0	0.0	0.0	0.0	0.0	8.54E-02	0.0	0.0	0.0	0.0	0.0	0.0
641610	3.12E-03	0.0	0.0	0.0	0.0	7.13E-01	0.0	7.63E-02	1.60E-03	3.76E-02	1.030	0.390
651610	1.16E-06	0.0	0.0	0.0	0.0	7.14E-01	0.0	0.0	0.0	0.0	0.275	0.003
661610	0.0	0.0	0.0	0.0	0.0	4.37E-01	0.0	0.0	0.0	0.0	0.0	0.0
641620	2.20E-08	1.000	0.0	0.0	0.0	1.07E-01	0.0	0.0	8.00E-04	1.73E-02	0.574	0.0
651621	1.54E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.130	0.0
651620	8.75E-05	0.0	0.0	0.0	0.0	2.82E 00	0.0	0.0	0.0	0.0	1.130	0.0
661620	0.0	0.0	0.0	0.0	0.0	9.53E-02	0.0	0.0	0.0	0.0	0.0	0.0
651631	1.65E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.80E-04	0.0	0.825	0.310
651630	2.96E-05	0.0	0.0	0.0	0.0	5.91E-01	0.0	0.0	1.80E-04	3.85E-03	0.825	0.310

NUCLIDE LIBRARY FOR AI REFERENCE OXIDE LMFBR

FISSION PRODUCTS

NUCL	DLAM	FB1	FP	FP1	FT	SIGNG	FNG1	Y25	Y28	Y49	Q	FG
661530	0.0	0.0	0.0	0.0	0.0	4.96E-01	0.0	0.0	0.0	3.85E-03	0.0	0.0
651640	8.37E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.20E-04	0.0	1.580	0.0
661540	0.0	0.0	0.0	0.0	0.0	1.21E-01	0.714	0.0	0.0	2.89E-03	0.0	0.0
661651	9.17E-03	0.0	0.0	0.0	0.0	5.48E-01	0.0	0.0	5.00E-05	1.16E-03	0.569	0.022
661550	8.30E-05	0.0	0.0	0.0	0.0	5.48E-01	0.0	0.0	0.0	0.0	0.531	0.027
671650	0.0	0.0	0.0	0.0	0.0	7.96E-01	0.0	0.0	0.0	0.0	0.0	0.0
661560	2.36E-06	0.0	0.0	0.0	0.0	1.39E-01	0.0	0.0	2.70E-05	6.55E-04	0.143	0.0
671561	1.83E-11	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.817	0.988
671660	7.16E-06	0.0	0.0	0.0	0.0	3.26E 00	0.0	0.0	0.0	0.0	0.739	0.0
681560	0.0	0.0	0.0	0.0	0.0	1.59E-01	0.010	0.0	0.0	0.0	0.0	0.0
681670	0.0	0.0	0.0	0.0	0.0	5.90E-01	0.0	0.0	0.0	0.0	0.0	0.0

SUM OF YIELDS OF ALL FISSION PRODUCTS = 2.00E 00 2.06E 00 2.02E 00

NUCLIDE LIBRARY FOR LMFBR

ACTINIDES

NUCL	DLAM	FB1	FP	FP1	FT	FA	SIGNG	FNG1	SIGF	SIGN2N	SIGN3N	Q	FG
HE 4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TL208	3.73E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.300	0.82
PB208	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PB212	1.81E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.299	0.60
BI212	1.91E-04	0.0	0.0	0.0	0.0	0.360	0.0	0.0	0.0	0.0	0.0	2.710	0.03
PO212	6.93E 03	0.0	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	8.940	0.0
PO216	4.62E 00	0.0	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	6.899	0.0
RN220	1.24E-02	0.0	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	6.396	0.0
RA224	2.20E-06	0.0	0.0	0.0	0.0	1.000	4.00E-01	0.0	4.00E-01	0.0	0.0	5.782	0.00
TH228	1.15E-08	0.0	0.0	0.0	0.0	1.000	4.00E-01	0.0	4.00E-01	0.0	0.0	5.525	0.0
TH229	3.01E-12	0.0	0.0	0.0	0.0	1.000	4.00E-01	0.0	3.00E 00	0.0	0.0	5.100	0.0
TH230	2.75E-13	0.0	0.0	0.0	0.0	1.000	4.00E-01	0.0	4.00E-01	0.0	0.0	4.767	0.0
TH231	7.52E-06	0.0	0.0	0.0	0.0	0.0	4.00E-01	0.0	3.00E 00	0.0	0.0	0.233	0.43
TH232	1.56E-18	0.0	0.0	0.0	0.0	1.000	4.44E-01	0.0	1.37E-02	1.20E-03	1.65E-05	4.080	0.0
TH233	5.23E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.420	0.0
PA231	6.76E-13	0.0	0.0	0.0	0.0	1.000	8.00E-01	0.0	3.83E-01	0.0	0.0	5.148	0.0
PA232	6.08E-06	0.0	0.0	0.0	0.0	0.0	3.60E-01	0.0	3.10E 00	0.0	0.0	0.560	0.40
PA233	2.93E-07	0.0	0.0	0.0	0.0	0.0	8.00E-01	0.512	3.80E-01	1.62E-03	1.65E-05	0.428	0.82
PA234M	9.87E-03	0.0	0.0	0.0	0.001	0.0	3.60E-01	0.0	3.10E 00	0.0	0.0	0.869	0.01
PA234	2.85E-05	0.0	0.0	0.0	0.0	0.0	3.60E-01	0.0	3.10E 00	0.0	0.0	1.700	0.81
U232	3.05E-10	0.0	0.0	0.0	0.0	1.000	3.77E-01	0.0	4.19E-01	1.95E-04	2.25E-06	5.414	0.0
U233	1.36E-13	0.0	0.0	0.0	0.0	1.000	3.96E-01	0.0	3.15E 00	2.48E-04	4.72E-07	4.909	0.0
U234	8.89E-14	0.0	0.0	0.0	0.0	1.000	4.50E-01	0.0	5.10E-01	5.40E-04	3.30E-06	4.856	0.0
U235	3.09E-19	0.0	0.0	0.0	0.0	1.000	2.92E-01	0.0	3.57E-02	1.12E-03	3.52E-06	4.681	0.0
U236	9.19E-16	0.0	0.0	0.0	0.0	1.000	5.35E-01	0.0	6.18E-01	4.88E-04	5.40E-06	4.573	0.0
U237	1.19E-06	0.0	0.0	0.0	0.0	0.0	2.60E-01	0.0	2.72E 00	0.0	0.0	0.362	0.26
U238	4.87E-18	0.0	0.0	0.0	0.0	1.000	3.01E-01	0.0	4.65E-02	1.12E-03	1.12E-05	4.268	0.0
U239	4.92E-04	0.0	0.0	0.0	0.0	0.0	2.60E-01	0.0	2.70E 00	0.0	0.0	0.400	0.0
NP236	8.75E-06	0.0	0.430	0.0	0.0	0.0	3.59E-01	0.0	3.15E 00	0.0	0.0	0.476	0.0
NP237	1.03E-14	0.0	0.0	0.0	0.0	0.0	7.65E-01	0.0	3.60E-01	9.75E-05	7.10E-07	0.0	0.0
NP238	3.82E-06	0.0	0.0	0.0	0.0	0.0	3.60E-01	0.0	3.10E 00	0.0	0.0	0.866	0.69
NP239	3.41E-06	0.0	0.0	0.0	0.0	0.0	8.28E-01	0.0	3.60E-01	0.0	0.0	0.500	0.32
PU236	7.71E-09	0.0	0.0	0.0	0.0	1.000	2.20E-01	0.0	1.40E 00	0.0	0.0	5.870	0.0
PU238	2.47E-10	0.0	0.0	0.0	0.0	1.000	2.24E-01	0.0	1.38E 00	1.95E-04	2.25E-06	5.587	0.0
PU239	9.00E-13	0.0	0.0	0.0	0.0	1.000	4.15E-01	0.0	1.89E 00	3.75E-04	4.35E-06	5.243	0.0
PU240	3.25E-12	0.0	0.0	0.0	0.0	1.000	5.18E-01	0.0	4.17E-01	2.62E-04	2.62E-06	5.255	0.0
PU241	1.69E-09	0.0	0.0	0.0	0.0	0.0	3.64E-01	0.0	3.17E 00	5.90E-04	5.90E-06	0.007	0.0
PU242	5.80E-14	0.0	0.0	0.0	0.0	1.000	3.60E-01	0.0	3.30E-01	5.50E-04	5.50E-06	4.980	0.0
PU243	3.87E-05	0.0	0.0	0.0	0.0	0.0	5.68E-01	0.0	2.03E 00	0.0	0.0	0.240	0.02
AM241	4.80E-11	0.0	0.0	0.0	0.0	1.000	9.90E-01	0.162	4.63E-01	3.75E-05	3.75E-07	5.630	0.0
AM242M	1.45E-10	0.0	0.0	0.0	1.000	0.0	4.03E-01	0.0	1.83E 00	1.39E-03	1.43E-05	0.048	0.0
AM242	1.20E-05	0.0	0.180	0.0	0.0	0.0	4.03E-01	0.0	1.83E 00	1.39E-03	1.43E-05	0.225	0.0
AM243	2.87E-12	0.0	0.0	0.0	0.0	1.000	5.55E-01	0.0	2.37E-01	0.0	0.0	5.439	0.0
AM244	4.44E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.536	0.0
CM242	4.92E-08	0.0	0.0	0.0	0.0	1.000	3.80E-01	0.0	4.20E-01	0.0	0.0	6.217	0.0
CM243	6.86E-10	0.0	0.0	0.0	0.0	1.000	4.00E-01	0.0	3.20E-01	3.75E-04	3.75E-06	6.161	0.0
CM244	1.21E-09	0.0	0.0	0.0	0.0	1.000	2.20E-01	0.0	1.40E 00	0.0	0.0	5.902	0.0